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SOVIET ATOMIC ENERGY

АТОМНАЯ ЭНЕРГИЯ
(ATOMNAYA ÉNERGIYA)

TRANSLATED FROM RUSSIAN

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SOVIET ATOMIC ENERGY

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FROM THE EDITORS

The present issue of our periodical, which comes out on the eve of the XXIV congress of the Communist Party of the Soviet Union, publishes materials reflecting the activities of the member-nations of the Council for Mutual Economic Aid ["COMECON"] in the development of the peaceful uses of atomic energy. Two events of note preceded the publication of these articles. A scientific conference centering around the topics "Nuclear power, fuel cycles, radiation materials science" held in October 1970, on the occasion of the Jubilee celebration of the 100th anniversary of the birth of V. I. Lenin, on COMECON initiative at Ul'yanovsk, and a month later the XIX session of the Permanent Commission of COMECON on the peaceful uses of atomic energy was held in Moscow.

The year 1970 marks ten years since the founding of this Commission, and a report by the chairman of the Commission A. Petros'yants dealt with the Jubilee celebration, as did remarks by the heads of all the delegations.

The Ul'yanovsk conference undoubtedly constituted a great event in the scientific and technical life of the countries of the socialist commonwealth. This is attested to by the statistics characterizing the scale of the conference. There were over 200 specialists at the conference from Bulgaria, Hungary, the German Democratic Republic, Poland, Rumania, the Soviet Union, and Czechoslovakia; the participants at the conference heard about 100 scientific reports.

At the first plenary session, which was held on October 5, the participants at the conference were welcomed by the first secretary of the Ul'yanovsk district committee of the Communist Party of the Soviet Union, A. Skochilov. The heads of delegations took the floor following this, to deliver their reports: R. Georgiev "On the development of nuclear power industry in the Peoples Republic of Bulgaria," D. Osztrovsky "The status of reactor and atomic power research in the Hungarian Peoples Republic," L. Heine "On some topics in applications of nuclear power," S. Andziejewski "Developmental outlook of the nuclear power industry in the Polish Peoples Republic," A. Petros'yants "Nuclear power in the USSR," K. Barabas (Czechoslovakia) "Fifteen years of collaboration with the USSR and developments in the field of nuclear power." The reports by L. Heine and K. Barabas are published in this issue; the report by A. Petros'yants will be published in one of the future issues of the periodical. Unfortunately, the editorial staff has not been able to get hold of the remaining reports by delegation leaders.

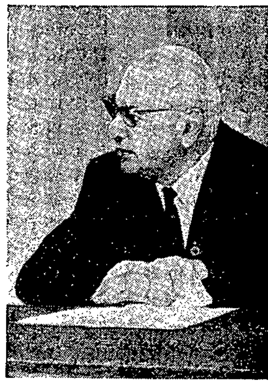
At the second (and concluding) plenary session, the chairman of the organizing committee of the Ul'yanovsk conference Professor O. Kazachkovskii took the floor to sum up the proceedings of the conference.

Most of the scientific and technical reports were presented at two panels on "Design and operation of nuclear reactors and nuclear power stations" and "Development, investigation, and radiation stability of structural materials and fuel materials for nuclear power stations." Twenty-five of these panel reports are presented to the reader in this current issue of the periodical, along with a list of all of the reports presented at the conference. Despite its wishes, the editorial staff has not been able to print all of the reports which it finds of great interest; but we hope that this gap will be filled by the publication of the entire proceedings of the conference sometime during the second half of 1971.

Translated from Atomnaya Énergiya, Vol. 30, No. 2, p. 90, February, 1971.

TENTH YEAR OF THE PERMANENT COMMISSION ON THE PEACEFUL USES OF ATOMIC ENERGY

REPORT OF THE PRESIDENT OF THE COMMISSION*



A. Petros'yants, President of the Commission, head of the Soviet Delegation

Ten years ago, in October, 1960, the Thirteenth Session of the Council of Economic Cooperation set up the Permanent Commission of the Council of Economic Cooperation on the peaceful use of atomic energy. Today in holding the nineteenth session of the Commission (or, allowing for an extraordinary session, the twentieth) we mark the tenth anniversary of its work. On the occasion of this festival it is appropriate to summarize our joint efforts.

The scientific and technical cooperation of the socialist countries in the peaceful use of atomic energy started in 1955 on the basis of bilateral agreements. The most vital aspect in the first few years of cooperation was the Soviet Union's scientific and technical aid to these countries in setting up research nuclear reactors, elementary-particle accelerators, and radiochemical and physical laboratories, and also (a particularly important feature) in training scientific personnel. Subsequent cooperation involved the development of individual scientific and engineering problems, new forms of nuclear-physics instruments, special equipment and methods of shielding technology, and help in organizing the production of radioactive isotopes and radioisotope apparatus.

In a very short time the countries of the socialist community assembled and initiated the use of installations and laboratories which constituted a basis for creating scientific centers of nuclear research. In setting up these centers and preparing them for operation more than 1000 highly-qualified Soviet specialists contributed their knowledge and experience, in addition to the national scientific staffs. Furthermore, over 3000 specialists and young scientists from the socialist countries have undergone education and training in the Soviet Union; many of these have defended dissertations and received Candidate's and Doctor's degrees.

In order to ensure wider and more systematic cooperation on a multilateral basis, 1960 saw the setting up of the Permanent Commission of the Council of Economic Cooperation on the use of atomic energy for peaceful purposes. The formation of the Commission constituted a reasonable and natural development of the first stage of cooperation, since at that time the countries constituting members of the Council were concerned with new problems associated with the use of atomic energy in the popular economy.

The successful solution of such vast scientific and technical problems as the industrial use of atomic energy required considerable appropriations, complicated and expensive equipment, a high level in the

*The Report and speeches are published in abbreviated form.

Translated from Atomnaya Énergiya, Vol. 30, No. 2, pp. 91-95, February, 1971.

development of interrelated branches of industry, and the creation of large staffs of scientific workers and specialists. Under these conditions the multilateral cooperation of the socialist governments, the division of labor, and mutual aid acquired a particular importance. Within the purview of the Commission for cooperation were included such important aspects as nuclear apparatus construction, isotopes and tracer compounds, radiation safety and shielding technology, and the use of radioisotope methods and apparatus in various fields of the popular economy. Following the proposals of a number of countries in 1965, the Commission organized cooperation in reactor science and technology and nuclear power.

During the last 10 years, the Commission for cooperation has completed a great deal of work in the development of nuclear-physics and radioisotope apparatus. The working group on nuclear apparatus construction has held 20 meetings and considered numerous scientific and technical and production problems; it has made 55 recommendations, including those on the standardization of detectors of ionizing radiations, electronic units and devices, dosimetric and radioisotope apparatus, and so on.

The Commission has devoted considerable attention to recommendations on specialization and cooperation in the production of nuclear-technology instruments; it has considered and made recommendations on the specialization of more than 70 types of equipment used in science, industry, agriculture, and medicine.

The following are examples of specialization in instrument production in countries of the Council for Economic Cooperation:

gamma-defectoscopic apparatus: Bulgaria, Poland, the USSR;

nuclear medical instruments: Hungary, the USSR;

nuclear-radiation counters: Poland and East Germany;

scintillators: Czechoslovakia and Hungary;

multichannel analyzers: Hungary and the USSR.

Following the decision of the twenty-third (special) session of the Council for Economic Cooperation, the working group on nuclear apparatus construction discussed the problem of new forms of cooperation: the creation of an International economic organization on scientific, industrial, and trade cooperation in the field of nuclear instruments "Interatominstrument" (IAI). In the extraordinary session in October 1969, the Commission incorporated this question in its "Complex program," and in the eighteenth session in June 1970 discussed the matter and recommended the creation of such an organization to the countries of the Council of Economic Cooperation.

It is well known that our Polish friends have organized the first session of the Preparatory Committee on the creation of Interatominstrument. We wish the Preparatory Committee all success in its enterprise.

One of the most important aspects in the activity of the Commission is the development of cooperation in producing and utilizing radioactive isotopes and ionizing radiations.

In all the countries of the Council of Economic Cooperation, radioactive isotopes and tracer compounds are now principally manufactured in research nuclear reactors and charged-particle accelerators. This has created conditions for close cooperation between the countries on developing methods for producing isotopes, regulating isotope production, specializing production techniques, unifying the main parameters and characteristics of the isotopes and tracer compounds, and also exchanging experience.

At the present time the catalog of radioisotope production in these countries embraces over 5000 preparations and sources of radiations, including some 800 types of isotopes and inorganic compounds, 900 organic tracer compounds, 300 radioactive medical preparations, 400 types of standard solutions and phosphors, 900 kinds of stable isotopes and specimens with enriched stable isotopes, and over 200 forms of closed radiation sources.

The Commission has accepted recommendations on specializing the production of 52 radioactive isotopes produced in the form of radioisotope products, some 300 inorganic preparations, and approximately 600 organic tracer compounds. The Commission has published a general catalog of radioactive isotopes and tracer compounds made in the countries of the Council of Economic Cooperation.

During the period of its work, the Commission has considered generalized material relating to the use of radioactive isotopes and nuclear radiations in various departments of the popular economy of the Council countries: in engineering and metallurgy, geophysics and mining, the oil and chemical industries, and building. On the basis of this information the cooperating countries are now exchanging industrial experience, technical documentation, instruments, and apparatus.

In the countries of the Council of Economic Cooperation, isotopes, tracer compounds, and radiation sources are used by 7000 undertakings and organizations, including 3600 industrial undertakings, 900 medical establishments, and over 1600 scientific-research institutes. Recently the annual growth in the number of undertakings and organizations using isotopes and tracer compounds, taken over all the Council countries has amounted to 10%, and in some countries to 30-40%.

In the fifteenth session of the Commission in November 1968, a "Report on the state and further development of work relating to the use of radioactive isotopes and nuclear radiations in the popular economy of the Council countries" was considered. The generalized data presented in this Report showed that in various branches of industry some 40,000 radioisotope instruments are now used, making a substantial economic contribution to these countries.

The eighteenth session, held in June 1970, considered and adopted a program for the further expansion of cooperation in isotope production between 1971 and 1975. This provided for the complex examination of problems relating to specialization and cooperation, unification and standardization, scientific and technical assistance, and the perfection of the information system.

In view of the formation of a working group on reactor science and technology and nuclear power in 1965 (there have been eleven meetings of this group), the Commission considered a number of important problems in this field.

The discussion of technical-economic data relating to the installation of 800 MW (electrical) atomic power stations with two VVÉR-400 units (water-cooled water-moderated reactors) and also the exchange of experience regarding the design, construction, and use of operative atomic power stations, all this has played a decisive part in the Council countries in the development of their national programs relating to nuclear power and to decisions as to the building of atomic power stations.

At the present time nuclear power is being widely developed in the Council countries as a new branch of power economy.

Bulgaria is providing for the introduction of a first 440 MW (electrical) power unit in 1974, and in the period up to 1980 the Republic proposes introducing atomic power stations to a total power of 2600 MW (electrical).

East Germany intends to bring its total atomic power station capability to more than 3000 MW (electrical) by 1980. Technical-economic calculations show that the cost of electrical power in the atomic power stations will be less than in thermal power stations based on brown coal.

Rumania is to build atomic power stations to a total power of 1800-2400 MW (electrical) by 1980, using reactors based on natural and enriched uranium.

In the Soviet Union a transition is taking place from the first experimental industrial atomic power stations to the construction of more powerful atomic power stations, chiefly in the European part of the Soviet Union. In some regions in which organic fuel is hard to obtain, the increase in power capacity as a result of the construction of atomic power stations in the next ten years will constitute a major proportion of the total increment in power capacity.

Czechslovakia is intending to erect atomic power stations with water-cooled water-moderated reactors, and is building atomic power stations with heavy-water reactors, the total power of these being estimated as reaching some 5000 MW (electrical) by 1985.

It is here appropriate to mention that the total power of all atomic power stations in the world is already more than 17,000 MW. A world-wide estimation of development indicated that in 1980 the atomic power station capacity will be over 300,000 MW and will contribute of the order of 15% of the power of all electrical power stations in the world.

The programs laid down in the majority of the Council countries and the speed of development of nuclear power envisaged in the 80's and 90's present our Commission and the engineers and scientists of the various countries involved with immense scientific and technical problems.

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The development of nuclear power on the industrial scale is a complicated and extremely important problem in popular economy; it demands the cooperative management of scientific institutes and undertakings in neighboring branches of industry and a deepening and more perfectly coordinated cooperation between the various countries belonging to the Council. In order to solve this problem the Commission is proposing to undertake a number of important measures in 1970-1972 together with other competent organizations of the Council of Economic Cooperation.

Cooperation is to be developed and perfected in the following fields:

in the field of nuclear medicine, including the development of methods and apparatus, particularly radioisotope scintillography;

in the field of developing radiative processes and installations of the experimental and industrial type, principally for the radiative sterilization of materials and the introduction of radiochemical processes into industry.

The introduction of radiative processes into the popular economy may be of considerable significance in improving the welfare of Mankind. By using radiative processes the period of storage of many products may be substantially extended, the products may be more easily transported, and losses may be significantly reduced, while preserving the quality almost unimpaired. All this is of great economic significance.

The agenda of the nineteenth session incorporates corresponding reports from the Hungarian delegation and the Temporary Working Group on radiation technology, including proposals for cooperation in these matters.

Unification and standardization play a major part in technological progress and in ensuring the quality, reliability, and long life of industrial products. Over the past ten years the Commission has carried out a great deal of work on the unification and standardization of the materials of nuclear technology and radioisotope production.

In summarizing the ten-year activity of the Permanent Commission of the Council of Economic Cooperation on the peaceful use of atomic energy, we notice that the principal aspect of this work has lain in scientific and technical cooperation and the coordination of scientific and technical investigations.

The plan for the coordination of scientific and technical investigations covering the period 1966-1970 includes more than 30 subjects relating to reactor technology and nuclear power, nuclear-physics instruments and radioisotope apparatus, isotope samples and radiation safety.

In order to provide a mutual exchange of scientific and technical information in relation to the results of the Commission's investigations, three exhibitions and seminars on the apparatus and instruments of nuclear technology have been held, in Moscow, Warsaw, and Leningrad, as well as 18 symposia and conferences.

Among these scientific and technical proceedings, special emphasis should be given to the scientific conferences and symposia organized and carried out by the delegations of the Council countries:

Hungary – on the control and monitoring of nuclear reactors and equipment in atomic power stations, on apparatus for activation analysis, on nuclear devices for medicine;

East Germany – on the safe disposal of nuclear waste, on the hydraulics and heat carriers of nuclear reactors, on parts used in nuclear instrument making;

Poland – on the physics and technology of research reactors, on new methods of producing radioactive isotopes;

USSR – on fast reactors, on water-cooled water-moderated reactors, on the further improvement of the characteristics of nuclear instruments, on nuclear power and nuclear materials science;

Czechoslovakia – on the processing and burial of radioactive waste, on the processing of irradiated nuclear fuel.

In the eighteenth session, the Commission approved a new plan for the coordination of scientific and technical investigations over the period 1971-1975. This plan provides for the further expansion and perfection of scientific – technical cooperation. According to this plan and also the annual plans of the Commission's work, many conferences and symposia will also be held in this five-year period.

In view of the expansion in the industrial use of atomic energy now taking place over the whole world, the General Assembly of the United Nations Organization, the International Agency for Atomic Energy (IAAE), the European Economic Commission (EEĆ), the United Nations Organization, and other international bodies are now studying questions as to the effect of atomic energy on the environment.

Questions as to the environment, including the prevention of the general contamination of the atmosphere, the soil, and the river basins by radioactive materials, inter alia, were considered by a special symposium of the International Agency for Atomic Energy in 1970.

A special conference of the European Economic Commission is to discuss these problems in Prague in 1971, so is a conference of the United Nations Organization in Stockholm in 1972.

The Commission's plan for 1971-1972 provides for developing a broad program of cooperation among the Council countries on radiation protection and shielding technology, including questions relating to the safety and reliability of operation of nuclear installations, methods and instruments, and also protective measures.

In the last two years the cooperation of the Council countries has been developing on the basis of the principles laid down in the twenty-third and twenty-fourth sessions of the Council of Economic Cooperation, which involved the participation of the leaders of the communist and workers' parties and the heads of governments.

These principles indicate the best ways for developing and perfecting scientific — technical and economic cooperation on a qualitatively new basis, by extending socialist economic integration among the countries of the socialist community and seeking new and more effective forms and methods of cooperation.

During the last ten years, our cooperation in the Commission has made great advances in establishing specific practical links and in the constant cooperative search for the solution of new problems. Our common course now lies in a further coming together and mutual assistance based on socialist economic integration; this serves the interest of each country in particular and the aims of the whole socialist community in general.

REMARKS BY HEADS OF DELEGATIONS

N. Ivanchev (Peoples Republic of Bulgaria)

Taking note, in his opening remarks, of the fact that the founding of the Council for Mutual Economic Aid was an act of vast political and economic significance, brought about by the objective requirements of collaboration between countries that had become liberated from capitalism and were launched on the pathway of socialist development, N. Ivanchev stated:

"Today, we mark the tenth anniversary of the founding of our Commission. This Commission consists of delegations with their responsible leaders, and a staff of highly skilled specialists, which enables us to discuss with competence, and solve, problems of major importance in the peaceful utilization of atomic energy.

Our delegation is in solidarity with the report delivered by A. Petros'yants. But permit me to draw up a rather brief balance sheet of the work of this Commission. Over its ten years of existence, the Commission has devoted close attention to collaboration in the development of technical progress, to exchange of scientific and technical documentation, to the training of cadres, and other similar tasks. In its sessions, the Commission has made decisions and offered recommendations to member-nations of the Council for Mutual Economic Aid in these matters. A great deal of work has been done on unitization and standardization of various instruments, equipment, and procedures. Unified rules and regulations have been worked out governing handling of radioactive materials, transportation and conveying of radioactive materials, and some of the topics appearing in the coordination plan for scientific and technical research have undergone impressive development. The first steps have been taken in specialization of some lines of equipment and instruments. But, as in any other area of progress, specialization confronts some unresolved problems. The chief prerequisite for improved effectiveness in this area is overcoming parallel operations in the production of some products, with the attendant poor concentration and production only in small batches. There should be further multifaceted coordination of specialization and cooperation in manufacturing, with due account given to the most complete utilization of the production resources and economic resources of the entire socialist system.

In line with the resolution adopted at the XXIII (special) session of the Council for Mutual Economic Aid and the 41st session of the Executive Council, our Commission has worked out a many-sided and comprehensive program.

In the name of the delegation of the Peoples Republic of Bulgaria, permit me to express our gratitude to the heads and members of all the delegations taking part in the Commission for the joint and fruitful activities which we have been able to engage in."

Translated from Atomnaya Énergiya, Vol. 30, No. 2, pp. 95-98, February, 1971.

D. Osztrowsky (Hungarian Peoples Republic)

Pointing out the fact that a scientific and technical revolution has unfolded over the past quarter of a century in Hungary, simultaneously with the construction of the new socialist society, and that a new branch of science and industry has begun to develop in that country, viz., the peaceful utilization of atomic energy, D. Osztrowsky enumerated the basic achievements of his country in that area. He went on to state:

"The delegation of the Hungarian Peoples Republic attaches great importance to the work of the Commission, particularly in the accomplishment of those problems which, in line with the resolutions of the XXIII and XXIV sessions of the Council for Mutual Economic Aid, are directed toward further improvements in economic and scientific-technical collaboration and socialist economic integration. One of the most important problems confronting us is that of laying down, through our joint efforts, the scientific-technical, production, and organizational prerequisites for accelerating the development and effective utilization on all levels of the national economy of the member-nations of the Council for Mutual Economic Aid of nuclear power on an industrial scale.

The work of the Commission over the past ten years, and the assistance rendered by the Soviet Union, have contributed to enhancing the role of atomic energy in the economies of the member-nations of the Council for Mutual Economic Aid. The nuclear power industry is acquiring steadily increasing significance in the power sector of the economy. Our collaboration, within the framework of the Commission, has made it a possibility for member-nations of the Council for Mutual Economic Aid to develop their nuclear power programs and to start work on the construction of nuclear power stations.

The development of nuclear power on an industrial scale requires such enormous economic and scientific-technical efforts that there will be a still greater need to deepen and expand collaboration still further in this area.

In proceeding to the accomplishment of these new and significant tasks, we can state with assurance that the ten years of successful activities on the part of the Commission have forged all the necessary prerequisites for the further development of collaboration between the member-nations of the Council for Mutual Economic Aid in the field of the peaceful uses of atomic energy."

"The ten years of activity by our Commission have been of tremendous importance for the German Democratic Republic. The German Democratic Republic, being a relatively small country, would not have been able to master the utilization of atomic energy through its own efforts alone. Consequently, taking advantage of this opportunity, I should like to express our gratitude to our Soviet comrades and friends, who have aided us in an effective manner, in a spirit of socialist solidarity, in bringing the level of peaceful uses of atomic energy up to the prevailing international level in such a short time. Thanks to the deliveries of the VVR-S research reactor, the U-120 type cyclotron, and the equipment for the Rheinsberg nuclear power station, as well as the opportunities to have students trained at Soviet institutes, the decisive scientific and material-technical prerequisites have been laid down in the German Democratic Republic for active collaboration on our part within the framework of the Commission.

The work of the Commission has contributed to a decisive degree to the joint development of a workable general line on the development of nuclear engineering and nuclear power in our countries. This joint development of crucial problems allows us the opportunity to use the resources of the member-nations of the Council for Mutual Economic Aid, in scientific and technical areas, to maximum advantage, and has allowed us to gain much time in the introduction of nuclear engineering and nuclear power into our national economy.

A. Rauh (German Democratic Republic)

In ten years of activities, we have been adopting, step by step, increasingly more effective forms of collaboration.

The following results of the work of the Commission have been of special importance to the German Democratic Republic:

the development of a comprehensive program to carry out the decisions of the XXIII Session of the Council;

discussion of the nuclear power station project with 800 MW water-cooled water-moderated reactors;

the report by the Executive Committee on nuclear power development progress in the member-nations of the Council for Mutual Economic Aid;

many years of successful work on the development and production of isotope-labeled compounds and sealed radiation sources of many types and sizes and competitive on the world market;

holding of expositions of nucleonic equipment, which has made a contribution to the acceptance of nuclear engineering techniques in all areas of the national economy.

The achievements and experience registered in the course of our collaboration prove that a new system of international relations, one corresponding to the essentials of the socialist structure, to the interest of each country in particular, and to the interests of the world socialist system as a whole, has been developed between the socialist states.

We are obliged to express our gratitude to all those who contributed to the successful work and resolution of important problems in the field of the peaceful uses of atomic energy, with the aim of strengthening and consolidating our countries, and also to those who are ready to join us in attempting to solve the problems of the future."

"The report made by the chairman of the Commission, as well as the remarks made by the leaders of the delegations of Bulgaria, Hungary, and the German Democratic Republic, seem to me to be an excellent reflection of the topic of the ten-year anniversary. I believe that it would be difficult to add anything essential to those remarks, and I wish only to thank comrade A. Petros'yants for his interesting report and his useful work as chairman of the Commission, and also to wish further success to all in their work."

Expressing his deep satisfaction that his first opportunity to participate in the work of the Commission as the assistant head of the Rumanian delegation coincided with the tenth anniversary of the Commission, A. Georgescu stated:

"We listened with great interest to the report presented by the chairman of the Commission, reflecting the important achievements of the member-nations of the Council for Mutual Economic Aid in the field of atomic energy, achievements which have been contributed to appreciably by the many-sided collaboration between the socialist countries, including Rumania.

In our country, broad opportunities for activities in the nuclear field were opened up as far back as 1956-1957, after the building of two important complex research facilities, the VVR-S reactor and the cyclotron, with assistance from the Soviet Union. A considerable portion of our research has centered around these two facilities. Our basic cadres, roughly 40% of whom have been through higher education,

S. Andzejewski (Polish Peoples Republic)

A Georgescu (Socialist Republic of Rumania)

underwent their training at these facilities, as well as at scientific research institutes and educational institutions of the Soviet Union.

At the present time there exist about 300 nuclear institutions and enterprises in Rumania, and production of several nucleonic instruments, radioisotopes, and labeled compounds has been worked out, as well as production of protective equipment for our own needs in the industry. In addition, we have worked out a program for further development of nuclear research, which includes building of nuclear electric power generating stations.

Unfortunately, because of unprecedented natural disasters which occurred this year in Rumania, our original schedules have now had to be revised (we have in mind a postponement of the program of building nuclear power stations, to go on the line before 1980), but this does not apply to the first nuclear power station, being built in collaboration with the Soviet Union; quite the contrary, we realistically expect to get this power station in operation even prior to 1978 as originally scheduled. That part of our program which deals with research of a fundamental nature, and with expanded applications of nuclear engineering and nuclear instrumentation, is not in for any change. Moreover, we consider that our metallurgical and machinery industry will have developed and mastered techniques and technology which will enable it to participate to a steadily increasing degree in the construction of nuclear power stations.

Fruitful and many-sided collaboration has been the rule throughout the ten years of work done through the framework of the Commission. Our country has striven, insofar as possible, to make its contribution to this cause. We have felt, in turn, the favorable effects of collaboration in all areas, and specifically in the area of nuclear power development.

After the XXIII (special) session and XXIV session of the Council for Mutual Economic Aid, activities within the framework of our Commission became intensified, and the methods of many-sided collaboration became improved.

Permit me in conclusion, to wish, in the name of the Rumanian delegation, new successes in the activities of our Commission, and in carrying out nuclear power developmental programs in our countries!"

The first secretary of the Embassy of the CSSR [Czechoslovak Socialist Republic] in Moscow, Z. Tluchor, stated:

"Permit me, on behalf of the Chairman of the Atomic Energy Commission of the CSSR J. Neumann, to congratulate you on the occasion of the tenth anniversary of the Commission. The Commission has completed some important assignments in the years of its existence. We must take note of the great work done in preparing cadres, in carrying out standardization programs, and programs promoting applications of radioactive isotopes and nuclear radiations, etc., and we must particularly stress the importance of collaboration in the field of nuclear power. Several successful conferences and symposia sponsored and organized by the Commission have been held. Important work establishing the prerequisites for widespread acceptance of nuclear engineering in all areas of the national economy of the member-nations of the Coucnil for Mutual Economic Aid has been done. The results acheived stand as testimony to the fact that the Commission has discharged the obligations imposed on it, and that collaboration between all the delegations has created the necessary preconditions for the Commission to successfully handle future problems. In the name of our delegation, I heartily thank the delegation of the USSR and all the remaining comrades for their international and disinterested collaboration."

LETTER OF WELCOME BY N. FADDEEV, SECRETARY OF THE COUNCIL FOR MUTUAL ECONOMIC AID

I wish to ask those in attendance to accept my congratulations on the occasion of the tenth anniversary of the Permanent COMECON Commission on the peaceful uses of atomic energy.

The establishment of this Commission in October 1960 was an important stage in the organization of collaboration between the member-nations of the Council for Mutual Economic Aid in the field of peaceful uses of atomic energy, one of the most crucial areas in scientific and technical progress.

With its activities based on the development of scientific and technical problems in the utilization of the achievements of atomic science and industry in different areas of the national economy, the Commission has carried out fruitful and extensive work over the past ten years in the development of many-sided collaboration in the field of nuclear power, and this has played a decisive role in the development of national programs on the construction of nuclear power stations by the member-nations of the Council for Mutual Economic Aid.

The Commission has devoted close attention in its work, to the development of recommendations on specialization and cooperation in the production of nucleonic instrumentation.

An important trend in the activities of the Commission has been the development of collaboration in the production and applications of radioactive isotopes and radiations in scientific research, in industry, in agriculture, and in medicine.

In the course of its activities, the Commission has performed some crucial work in developing unitization and standardization of nuclear hardware and radioisotope products, and has accepted over a hundred recommendations dealing with standardization.

Scientific and technical collaboration within the framework of the Commission has been a factor rendering great assistance to the various countries in establishing and developing their own respective atomic science and industry, in the organization of scientific centers for nuclear research and for the training of scientific cadres.

Over the past ten years, collaboration between member-nations of the Council for Mutual Economic Aid within the framework of the Commission has been steadily moving forward, on the path to establishing concrete practical liaisons and a constant search for solutions to new problems and tasks.

In recent years, the activities of the Commission have been centered on fulfilling the resolutions of the XXIII and XXIV sessions of the Council for Mutual Economic Aid, which set down guidelines for further development and improvement of scientific-technical and economic collaboration on the basis of socialist economic integration, featuring utilization of more effective forms and methods of collaboration.

The Commission has achieved positive results thanks to the active collaboration of leaders of delegations and of a large number of scientists and engineering and technical workers.

I wish all the participants of the Commission new successes in their work.

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SCIENTIFIC-TECHNICAL CONFERENCE ON ATOMIC POWER, FUEL CYCLES, AND RADIATION EFFECTS IN MATERIALS: PLENARY PAPERS

SOME PROBLEMS IN THE USE OF NUCLEAR POWER

A. Rauh and L. Heine

It is now clear that further electrification of our countries during the next decade will be based more and more on atomic power. We discuss below some of the problems of predicting the development, starting from the premise that we can successfully solve these problems only with socialistic collaboration.

Nuclear power from nuclear fission and thermonuclear reactions provides a new primary power source which will be available practically indefinitely. In contrast to other methods, the generation of electric power by nuclear fission can lower the power cost, decrease the manpower needs, improve industrial and everyday conditions for power plant personnel and people living in the vicinity, improve the purity of the biosphere, and ensure a more reliable power supply, particularly during the winter months, since the operation of an atomic power plant is less dependent on climatic conditions.

At the present time most of the power in the German Democratic Republic (GDR) is generated by the combustion of lignite. A radical change in the pattern of power sources is contemplated for the future, however. The conversion to nuclear power in the GDR is necessary also because the cost of mining lignite is increasing.

For the above reasons exceptional importance is attached to the development of atomic power in the GDR. Up to 1980 we propose to prepare for the transition to the generation of electric power on an industrial scale by atomic power plants and to start up the first large scale atomic power plants. One of these is the Nord atomic power station now under construction on the Baltic seacoast near the town of Greifswald.

After 1980 increased power demands will have to be met largely by atomic power plants. The development of nuclear power, however, cannot be discussed apart from the development of the power industry as a whole. This development is related to the pattern of power demands, to the development of investment funds and working capital, to the utilization factor and the possibility of operating power plants at various load factors, to the automation of the whole power supply system, and to the combination of atomic power plant operating conditions and local conditions.

The problem which has decisive importance for the development of nuclear power is the lowering of atomic power plant costs, and the essential factor here is the increase in the power of units and individual plants – the enlargement of equipment. On the basis of international experience it can be predicted that doubling the power of a unit under present conditions will lower costs by 10-20%. In considering questions of long-term development it should be taken into account that the power of atomic power plant units will be increased as installed capacity increases. Only such an approach holds the expenditures for experimental-structural, design, and structural-assembly operations, and the number of structural items to a level justifiable by the national economy. This applies also to the total power of the atomic power station. The combined power grids of our countries will be stable enough to permit placing atomic power plants at sites which will allow the necessary increase in unit power.

An essential prerequisite for lowering costs is the development of optimized plans and designs for the mass production of standard atomic power plants using efficient modern construction and assembly technology. In addition operating conditions must be achieved which will lower the servicing costs, permit an optimum ordinary maintenance routine, and at the same time ensure a high utilization factor. Since we must seek the lowest construction costs, preparatory work, especially design, is of great importance.

Soviet scientific research collectives have made advances in the field of reliable and high-power fuel element development which have set the scientific and technical world standards.

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The lowering of fuel costs requires not only a steady increase in the overall efficiency of atomic power plants, but also a close cooperation between our countries in developing a highly productive fuel industry to ensure a reliable and economical supply of fuel elements and assemblies to atomic power plants.

The varied and complex problems posed greatly exceed the scientific, technical, and economic capabilities of a single country such as the GDR.

We can develop nuclear power in our countries only if we can organize long-lasting close cooperation between our countries, particularly with the USSR, at all stages of the material procurement process, and achieve an appropriate division of work and specialization in the development and construction of systems and equipment subassemblies.

A complicated but important problem of reactor strategy arises in connection with the development of nuclear power, i.e., the choice of the type of atomic power plant for further development. We consider a two-type strategy the most effective, and foresee the use of pressurized water reactors in the first stage and sodium-cooled fast-breeder reactors in the second.

Accordingly the initial units at the Nord atomic power plant will have 440 MW water-cooled water-moderated reactors developed in the USSR. The first units should be started up in the course of the next Five-Year Plan. In the second half of the nineteen seventies we intend to turn to units with 1000 MW reactors. Thus in the seventies we will install a highly developed type of atomic power plant to ensure high reliability of operation and a high utilization factor to permit us to achieve favorable technical and economic showings. All this will be a logical development of the chain begun with the construction of the atomic power plant at Reinsberg.

Water-cooled water-moderated reactors have some disadvantages too, such as the poor utilization of nuclear fuel (0.5%, and no more than 2% for a closed cycle), and the fact that it is difficult and expensive to increase their power significantly. Because of the limited supplies of natural uranium these drawbacks necessitate the introduction of industrial-scale fast-breeder reactors as quickly as possible. On the basis of the high level of technical work in this direction achieved in the USSR we assume that reactors of this type will be introduced in other countries of the CMEA GDR during the first half of the nineteen eighties. These reactors significantly increase the efficiency of electric power plants, lower the fuel cost, and solve the fuel problem by their high utilization of uranium. Therefore the problem of producing an effective fuel economy, in particular of a closed fuel cycle, becomes especially important. Only this enables the economic advantages of breeder reactors to be completely realized. Whereas the operation of atomic power plants with water-cooled water-moderated reactors will lead to a fuel economy in our countries based on contractual cooperation with the USSR in obtaining fresh, and if it is expedient, reprocessed spent fuel, the use of breeder reactors will of economic necessity require the creation of fuel cycle systems in other countries as well as in the USSR.

In view of the considerable increase in installed electric capacity at atomic power plants it is to be expected that at least some of the fast reactor atomic power plants under construction will be started up on enriched uranium. Therefore cooperative studies directed toward a reduction of the demands for natural uranium to a minimum and toward the removal of the peak requirement for enriched output predicted for 1990 must be performed in time. Such studies are very complicated since they must examine the whole atomic power plant system. In developing an atomic power plant system the fuel economy requires taking account of a new functional relation. This emphasizes the complex nature of nuclear power problems.

Another important atomic power plant problem is the choice of site. In contrast with conventional power plants atomic power plants have the advantage that in principle they are independent of the location of the fuel supply. One can then try to choose the power plant site on the basis of power demands. The demonstrated safety of atomic power plants already permits their construction near high power demand centers. Site selection for an atomic power plant is limited mainly by the adequacy of the water supply for cooling the turbine condensers. This applies both to once-through cooling and to cooling with cooling towers. The development of air cooling systems in the future will permit the choice of power plant site to be made from considerations of increased power plant efficiency. To this end cooperative research must be directed toward using this method in connection with the enlargement of atomic power plant units and to a significant improvement of their technical and economic showings. It is also necessary to consider methods of distributing electric power in a combined grid, the properties of soils and ground water, meteorological conditions, conditions for connecting the building area with the transport system, problems of producing everyday items, connecting small residential areas with already assimilated parts of the territory, and minimum use of agricultural areas. The final solution is adopted as the result of economic optimizations directed toward minimizing the costs which depend on local conditions, taking account of all interacting factors. The first studies showed that there are many places in the GDR of equal economic value as sites for power plants, guaranteeing them adequate power.

In conclusion we take note of a problem which we believe will become more and more important as time goes on.

The national economy requires a large amount of useful energy in the form of heat, mostly hot water and steam. The replacement of these energy sources by electricity will take place gradually and in addition this transition is essentially dependent on lowering the cost of electrical energy. Therefore the question of using atomic energy for heating is of ever increasing interest. Experiments have shown that the combined generation of electrical and thermal energies by an atomic power plant is economically competitive with other methods of producing thermal energy. These experimental data are applicable to the types of power plants being designed or constructed at the present time in the USSR. A question arises in connection with large-scale atomic power plants constructed in regions where there is a large demand for thermal energy. In this case special problems arise related to the choice of suitable locations ensuring minimal extension of central-heating networks, the presence of an adequate supply of cooling water, and the creation of reserve power to ensure the reliability of the heat supply.

In this paper we wanted to touch upon a number of problems which must be solved in the forthcoming Five-Year Plan. No matter how difficult and complex these problems may be we will solve them, and thereby contribute to the economic and political solidarity of our countries.

FIFTEEN YEARS OF COOPERATION WITH THE USSR IN ATOMIC TECHNOLOGY, AND THE PROSPECTS OF THE DEVELOPMENT OF NUCLEAR POWER IN CZECHOSLOVAKIA

K. Barabas

The use of atomic energy in order to produce electrical power has opened up important new prospects for satisfying power requirements; the importance of this lies in the fact that Czechoslovakia's own resources of mineral fuel will be exhausted in 50-60 years time. On April 23, 1955, the first agreement between the Soviet Union and Czechoslovakia in relation to the setting up of an Institute of Nuclear Studies in Rzez near Prague was accordingly signed. The Soviet Union guaranteed the main equipment (the water-cooled water-moderated reactor and the cyclotron) and helped train Czechoslovakian specialists. In this way we were gradually able to assimilate the theoretical and practical bases for the peaceful use of atomic energy.

Of considerable importance in the cooperation between Czechoslovakia and the Soviet Union in the peaceful use of atomic energy was the agreement signed in March 1956 regarding the assistance of the Soviet Union in designing and building the first Czechslovakian atomic-power station. This station has a power of 150 MW and uses a heavy-water reactor cooled with carbon dioxide, the fuel being in the form of natural metallic uranium; at the present time the station is ready to go into service, the physical initiation being planned for June 1971. All the necessary measures are being taken in order to introduce one of the three 50 MW turbines into service by the end of September 1971. The delay in constructing our first atomic-power station is mainly due to pessimism regarding nuclear power between 1958 and 1962, as a result of which the installation of the first atomic-power station A-1 was almost halted for four years. Another reason was the fact that the A-1 was designed for a power of 150 MW, this power being large for a prototype. It should nevertheless be pointed out that the erection of the A-1 station has enabled us to educate our own staff of scientific and technical specialists and set up the basic conditions in the engineering industry for producing the technological eqiupment required in nuclear power.

In view of the fact that there will be a substantial deficit to cover in the energy balance of the country, even in the 1971-1980 decade, Czechoslovakia is now erecting atomic-power stations with reactors of the VVER-440 (water-cooled water-moderated) type, which have already been tested in service. On the recommendation of the Czechoslovakian Commission on Atomic Energy, the Czechoslovakian Government has requested the Soviet Government to erect two atomic-power stations by 1980, each with two reactors of the type mentioned. A corresponding agreement was signed by the two Governments on April 30, 1970. According to this agreement, the Soviet Union will supply the set of equipment for the whole of the first circuit, while Czechoslovakian factories will supply the equipment for the second circuit. In July 1970 the President of the State Committee on the Use of Atomic Energy of the USSR, A. M. Petros'yants, and the President of the Czechoslovakian Commission on Atomic Energy, Jan Neumann, signed an agreement on the carrying out of cooperative scientific and technical work in the development of water-cooled water-moderated reactors and fast-neutron reactors of high specific powers.

TABLE 1. Forecast of the Structure of Power Sources (%)

Source of power	1950	1960	1970	1980	1990	2000
Atomic-power stations	—	—	—	3,3	17,9	40,4
Oil	1,6	7,1	20,1	29,8	31,9	25,4
Natural gas	—	2,9	3,3	7,6	6,1	4,3
Brown coal	41,3	47,1	43,5	34,0	27,5	19,0
Ordinary coal	55,9	40,6	30,3	23,1	14,1	8,6
Imported raw materials and hydroelectric power	1,2	2,3	2,3	2,2	2,5	2,3

During the 15 years of cooperation with the USSR in the field of atomic technology, more than 1000 Czechslovakian specialists have been trained in Soviet institutes in Moscow, Obninsk, Khar'kov, Melekess, and the

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Novo-Voronezh and Beloyarsk atomic-power stations. All this bears witness to the fruitful cooperation between the USSR and Czechslovakia.

Regarding the prospects of the development of nuclear power in Czechoslovakia, it should be noted that over a number of years we have been holding discussions with specialists in the field of classical power in order to advance the development of nuclear power. In making economic comparisons, some of these specialists have tended to separate nuclear power out from the overall economical estimate; this has rather concealed the fact that the results of research and development in the field of nuclear power concern every branch of technology and popular economy.

Without accepting this feature of atomic power it is impossible to understand its importance in the popular economy and in the scientific and technical revolution upon which we are now entering.

On the other hand, it became absolutely vital to further the concept of the development of nuclear power up to 1980, not only from the point of view of overcoming the deficiency in the energy balance, but also from the point of view of creating conditions for research activity in the present decade (1971-1980) together with the wider development of industrial and construction work in connection with the installation of atomic-power stations, which after 1980 will provide almost the whole of the growth in electrical power needed. All the work which has already been done, together with forecasts of the development of power in Czechoslovakia, emphasize the necessity of the extensive construction of atomic-power stations (Table 1).

The structure of the use of power sources in Czechoslovakia differs substantially from that of all the developed countries, primarily in the high relative use of coal and the low relative use of electrical power. Furthermore, at the present time, our specific consumption is some 6 units of comparison fuel per inhabitant per annum, which gives Czechoslovakia third place (after the United States and Canada) in the world; at the same time this indicates the extremely high consumption of power per unit national revenue.

Of course, brown coal, with its high ash and sulfur content, presents problems from the point of view of preserving the environment. Thus, for example, in Prague 80% of the inhabitants suffer from environmental contamination. In certain parts of Prague the contamination is four to seven times more than the norm, which in Czechoslovakia equals 150 tons of solid fall-off per 1 km^2/year. In Brno the norm is exceeded by a factor of 10. Roughly the same applies to other cities, for example, in Pilsen, Bratislava, and Ostrava, where the solid fall-out reaches 650 tons/km^2/year.

All this indicates the vital importance of building atomic-power stations. On the basis of concepts developed by the Czechoslovakian Committee of Atomic Energy, forecasts for the period up to 1990 indicate the following introduction of atomic-power station capacities:

	1980	1985	1990
Atomic-power station capacity, thousand MW (electric)	1.7	5.0	12.0
Proportion of total capacity, %	7.7	17.1	29.4
Proportion of total power production, %	12.0	26.1	41.8

If the erection of thermal-power stations is to be limited from the point of view of protecting the environment, the capacity of the atomic-power stations will have to be raised by 2500 MW in 1985 and 3500 MW in 1990.

In addition to the construction of atomic-power stations for power purposes, it is also intended to construct atomic heat and power centers, which in addition to electrical power will also produce heat community and industrial needs. Estimates indicate the economic viability of atomic heat and power centers. The best place in Czechoslovakia to build the first of these is Brno, which has a well-developed heat-supply system; furthermore this city particularly needs environmental improvement.

As noted earlier, doubts as to the feasibility of carrying out the program of erecting atomic-power stations with heavy-water reactors in the set time, together with the growing need for power, have compelled us to consider the necessity and desirability of importing atomic-power stations already tested on the industrial scale. This necessity arose from the Government's insistence on comparing different types of reactors, especially those of the heavy-water type.

The Czechoslovakian Committe of Atomic Energy gave preference to the VVER-440 reactor and recommended to the Government that discussions should be initiated with the USSR regarding the supply of two atomic-power stations (each with two VVER-440 reactors) to be put into service by 1980. A corresponding agreement has now been reached between Czechoslovakian and Soviet Governments.

On reconsidering our preference for the construction of atomic-power stations with VVER-440 reactors at the present time, I am convinced that this decision was a right one politically as well as from the economic point of view. The decision opens up new prospects for extensive industrial cooperation between member countries of the Council of Economic Cooperation, chiefly with the USSR, and in my own opinion will assist fundamentally in solving the problem of economic integration within the framework of the Council. Our decisions are also justified by current circumstances in other countries (for example, in France and Sweden, in which the program of heavy-water reactors is passing through a stage of reappraisal, and also in Canada, where difficulties have arisen in the supply of heavy water).

In future the installation of atomic-power stations in Czechoslovakia will proceed in close cooperation with the USSR and the other member countries of the Council for Economic Cooperation. This kind of solution to problems of developing nuclear power in the socialist camp will strengthen the political and economic basis for the whole of our socialist system.

BASIC OPERATING CHARACTERISTICS OF THE REACTOR INSTALLATION IN THE SECOND UNIT OF THE NOVO VORONEZH NUCLEAR POWER STATION*

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Part 1. Startup of the Second Unit of the Novo Voronezh Nuclear Power Station

The second reactor power unit of the Novo Voronezh nuclear power station was brought into operation in December 1969. The second unit shares much in common with the first unit in terms of basic heat power arrangements, engineering design solutions, and types of basic process equipment. This second unit constitutes in its own right an entirely new stage in the development of nuclear power stations incorporating VVER reactors. Below we cite the basic parameters of the second unit:

Reactor thermal power	1320 MW
Electrical power output	365 MW
Number of subloops in first loop	8
Primary-loop pressure	105 kg/cm^2
Average temperature of primary-loop coolant	260°C
Pressure of dry saturated steam produced in steam generators	32 kg/cm^2
Number of turbogenerators	5
Duration of first campaign	195 effective days

The composition of the first fuel loading, and the nuclear physics characteristics of the first loading, are entered in Tables 1 and 2.

The almost 80% rise in thermal power output over that of the first unit in this power station was achieved while using a reactor pressure vessel of the same diameter, through improved utilization of the reactor core: this involved the use of fuel elements of smaller diameter in the working fuel assemblies (9.1 mm instead of the previous 10.2 mm), and smoothing out nonuniformities in the power generation pattern throughout the core. The combined effect was to increase the irradiation level in the core.

The capacity of the steam generator units was increased in a similar fashion: U-tubes of smaller diameter (ϕ 16 × 1.4 mm instead of the previous ϕ 21 × 1.5 mm) were used in the vessel of a steam generator of the same diameter as that installed on the primary loop. This measure increased the surface available for heat transfer by 40%, and increased steam capacity by a corresponding amount.

Of the fundamentally new engineering solutions adopted in the design of the second unit, in contrast to those adopted in the first unit, we must single out the use of a self-propelled carriage positioning the control rod assemblies for scramming the reactor; blowdown purging of the primary-loop water through ion-exchange columns; compensation of pressure fluctuations in the primary loop by means of steam pressurizers; the use of a "dry" reactor refueling system and a heat-sensitive nut-runner to seal the reactor

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TABLE 1. Composition of First Fuel Loading

Uranium enrichment, %	Number of fuel assemblies	
	working	ACRC
1,0	36	19
1,5	51	15
2,0	93	27
3,0	96	12
Total	276	73

Note: ACRC denotes fuel assemblies incorporated in a system of automatic control and compensation of excess reactivity.

top cover; monitoring the performance of the steam generating equipment with the aid of IV-500 data-processing machines; cooldown of the reactor and removal of residual afterheat by natural circulation through the primary loop and forced circulation of water through the secondary loop.

The performance of the first unit during the startup of the second unit facilitated many of the operations involving blowdown and run-in of the equipment. For instance, all the turbogenerators in the second unit were tested in operation at power levels of 40 to 60 MW, using steam from the steam generators in the first unit, thereby making it possible to carry out startup operations independently in the reactor division.

The startup operations in the reactor division of the nuclear power station included the following operations: blowdown of the primary and secondary loops, and also of all the ancillary systems, after installation had been completed; blowdown of the primary loop hot and loading of the subcritical zone; thoroughgoing adjustments of the control rod system; inspection of reactor innards (equipment within the pressure vessel) and of the primary-loop equipment after hot run-in; loading the core for the first campaign; physical startup of the reactor, bringing the reactor up to 40% power level, on-power overall testing of equipment and process lines for power performance; gradual raising of reactor power to rated output level.

These stages took place during the startup of the first power unit of the Novo Voronezh nuclear power station as well. But these operations were completed in much shorter periods during the startup of the second unit.

Blowdown of Ancillary Systems. Blowdown of various reactor systems after installation was carried out immediately after installation of each separate system had been completed. The execution of these operations was expedited by the fact that it was now possible to feed chemically desalinated water in sufficient quantities from the first unit systems.

The clearly defined interrelations between the operating personnel and the personnel involved in installation and rigging work made it possible to eliminate shortcomings as they were discovered, to assemble in place temporary structures to aid in blowdown and run-in of the reactor ancillary systems, and to bring the entire system to working order immediately after the blowdown operations had been completed.

In summary, all of the ancillary systems of the primary loop were ready for service before the circulation subloops and the reactor has been washed down.

Pressurizing of the Primary Loop. Preliminary pressurizing of the primary loop, at 150 kg/cm^2 pressure, was carried out in order to shorten the time required to complete the startup and adjustment operations (this preliminary pressurization did not involve the electrically wired part of the main circulation pump [GTsN], the regular top units, or the in-pile reactor devices). The scroll of the main circulation pump was sealed by temporary covers, while the reactor pressure vessel was sealed by a test-stand cover.

The reactor pressure vessel was heated up to 100°C by feeding through steam under 5.3 kg/cm^2 from the main steam header of the first power unit.

Primary-Loop Blowdown. Before blowdown was begun, a metallic structure was set up inside the reactor pressure vessel to separate the interval volume of the reactor into two parts. This made it possible to blow down all the circulation subloops by using a single pump. Coolant was circulated along the following path (see diagram, Fig. 1): the head built up by the GTsN-12 pump fed coolant under the reactor pit, and from there through the cold branch of loop 12 and through the steam generator unit 12 to the intake of the GTsN-12 pump. The hydraulic resistance in this coolant path allowed the GTsN pump to operate at rated performance. The GTsN-12 pressure drop was 5 kg/cm^2 when the temperature of the primary-loop water was 32°C, and was 6.1 kg/cm^2 when the temperature of the primary-loop water was 180°C.

The reactor pressure vessel water was warmed up and kept at a stable temperature during the blowdown operation by supplying feedwater from the secondary loop of the first power unit into the second-unit reactor vessel via the test-stand cover rig; this feedwater entered at temperatures to 195°C.

TABLE 2. Reactivity Excess and Effect of Reactivity Changes

Parameters	ρ, %	$\Delta\rho$, %
Reactivity excess:		
at 20°C	13.77	-
at 260°C	10.16	-
Temperature effect (20-260°C)		-3.61
Power effect (0-100%)	-	-1.61
Stationary Xe^{135} poisoning*	-	-2.80
Stationary Sm^{149} poisoning*	-	-0.58
Reactivity excess to compensate burnup*	5.17	-
Effectiveness of ACRC system:		
at 20°C	-	20.61
at 260°C	-	26.71
Initial subcriticality of core:		
at 20°C	6.73	-
at 260°C	16.55	-

* At 100% power.

The next subloop to be blown down was connected up without shutting off the GTsN-12 pump. Blowdown was carried out at pressures of 39-44 kg/cm^2 in the reactor, and the pressurizers filled up completely with water. Since the primary loop was connected to the secondary loop of the first power unit in operation by means of a 50 mm diameter pipe, the primary-loop pressure corresponded to the head developed by the feedwater pumps in the first unit. The subloop blown down was disconnected from the rest of the primary loop by means of shutoff gate valves; the pump stator assembly was assembled in the scroll of the GTsN scroll. The primary loop was blown down before all the GTsN pumps had been completely installed, and this meant that the startup of the second power unit proceeded about two months ahead of schedule.

Hot Run-In of Primary-Loop Equipment. The subcritical core was assembled from regular working assemblies in the control and protection system. This was backed up by a loading of 76 boron absorbing inserts to replace working assemblies, making it possible to raise up all 73 regular control and protection assemblies when required to check out and readjust the system.

The regular cover of the reactor, with all the control and protection instrumentation in place, was installed and sealed after the subcritical core had been loaded (this took from October 2 through October 28, 1969). The system of control mechanisms was first adjusted and checked out while the cover was in the inspection pit. Hot run-in of the primary-loop equipment lasted from November 4 through November 17, 1969. During that period the performance of all the GTsN pumps was checked out under different sets of operating conditions (with from one to eight GTsN pumps working at once); the primary-loop temperature ranged from 60° to 220°C during the checkout. During that time the control rod system and the system for monitoring and automatically regulating the primary loop were run through final adjustments, and the performance parameters of the loop equipment were also measured.

The actual hydraulic resistance of the primary loop turned out to be less than predicted. This made it possible to attain the design coolant flowrate with seven subloops operating (the design flowrate was 48,000 m^3/h when the delivery head of the GTsN pumps was 5.2 kg/cm^2, and a flowrate of 49,000 m^3/h was attained with seven subloops operating when the delivery head of the GTsN pumps was 4.3 kg/cm^2, converted to the temperature 250°C).

With eight subloops operating, the coolant flowrate through the reactor reached a level of 53,000 m^3/h (converted to design parameters), or 8.3% above the design rating.

As a result of the hot run-in, it was found that it is sufficient to have seven circulation subloops of the steam loop operating in order to reach the design power output level of the core and the required steam capacity of the power unit.

A decision was made to operate the second power unit at rated output level with seven GTsN pumps in operation, and with the eighth subloop held in reserve.

Inspection of In-Pile Devices. After the primary loop had been run in hot, the reactor cover was removed, the core was unloaded, the in-pile devices were withdrawn, and the steam generator headers were opened up. Inspection of the equipment disclosed no serious problems or flaws of any kind.

Loading of the Regular Core. The regular core was loaded after careful inspection of the in-pile devices, and after checking out all problems brought to light during the hot run-in of the primary loop. After the core had been loaded, the refueling system was tested.

Physical Startup of the Reactor. The physical startup was controlled from the control panel of the power unit. Regular control rod machinery, regular instruments for monitoring neutron dose rate, with

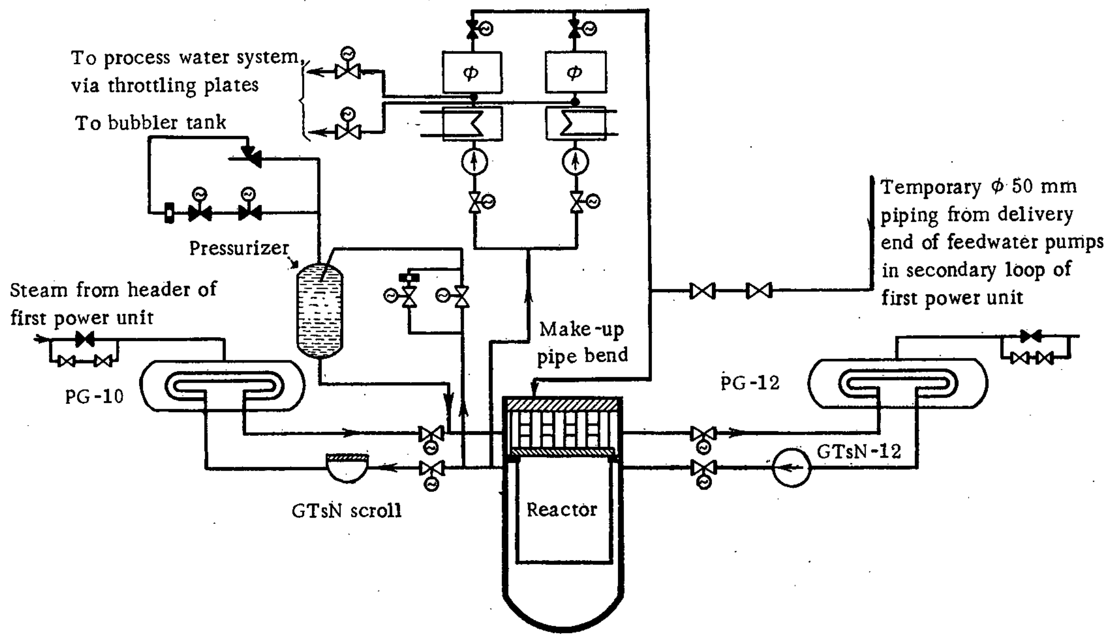

Fig. 1. Flowsheet for hot blowdown of primary loop with one GTsN-12 pump on stream and one subloop connected up.

the addition of a sensitive galvanometer with preamplifier, and two scalers, were used. All the transducers in the startup equipment were placed in regular channels of the biological shielding tank.

One interesting feature of the physical startup of the second power unit was the fact that a special module of transducers was not installed in the reactor core. This eliminated any distortion of the physical characteristics of the core, while making it possible to suspend the array of control rod assemblies with greater precision.

More precise neutron physics characteristics of the core were obtained during the physical startup and by operating the equipment at low power, further tests and checkouts were run on the performance of the control rod system, on the monitoring and measuring instruments and equipment, interlocks, pressure-tightness of the primary loop, reliable electrical supply circuits, the performance of the GTsN principal circulation pumps, and ancillary systems at the design pressure and at the water temperature 240°C.

Synchronization of the first turbogenerator in the second power unit with the power grid was achieved on December 27, 1969.

Power Startup. A comprehensive checkout of the systems and equipment was carried at 500 MW reactor thermal output level. At that output level the performance of the turbogenerators at rated parameters was tested, adjustments were made in the original water and chemical conditions of the primary loop, the heat source distribution through the core was investigated, and studies were made of the power effect and poisoning effect of the reactor, and of the stability of the power unit performance as part of the power grid.

The output power of the second-unit reactor was checked out at the successive levels: 5, 38, 50, 60, 75, 80, 90, and 100%. While operating at a fixed power output level, the reactor was run through various experiments designed to obtain more precise dynamic characteristics of the performance of the primary and secondary loops. A procedure was worked out at the same time to maintain the specified water conditions, and operation of the primary loop with boron coolant at a boric acid concentration to 3 g/kg, starting with 500 MW output, was mastered.

The water and chemical conditions of the second power unit did contrast with those of the first unit. In contradistinction to the first power unit, where a quadruple-effect evaporator plant is employed to purify the coolant, two ion-exchange filters, one working and one in reserve, are employed in the second unit (KU-2 cation-exchange resin in ammonia form for the cation-exchange filters, and AC-17-8ChS grade anion-exchange resin in OH-form for the anion-exchange filters).

TABLE 3. Graph of Power Output Levels Reached

Thermal power, MW	Percentage rated power	Electrical power, MW	Date power raised	Date operation ended at specified power level	Number of hours operated at specified power level	Number of effective days at specified power level
500	38	130	Dec. 29, 1969	Jan. 24, 1970	615	9,76
660	50	182	Jan. 27, 1970	Mar. 10, 1970	802	27,3
795	60	220	Mar. 10, 1970	Mar. 24, 1970	347	35,84
990	75	275	Mar. 24, 1970	Mar. 27, 1970	90	38,48
1055	80	290	Apr. 1, 1970	Apr. 9, 1970	216	44,62
1190	90	325	Apr. 9, 1970	Apr. 14, 1970	102	50,14
1320	100	365	Apr. 14, 1970			

The following water conditions were established during the period of initial reactor operation without boron control:

	Rating	Actual values
Hydrogen, Nml/kg	30-60	50-90
Oxygen, mg/kg	0.02	0.001-0.02
Chlorides, mg/kg	0.05	-
Ammonia, mg/kg	10-30	8-20
Corrosion products, mg/kg	0.1	0.1
pH	10-10.5	9.7-10.2
Activity A, Ci/liter	10^{-3}	$2 \cdot 10^{-5}$

The water conditions were maintained through the operation of the ion-exchange filters and metering of ammonia, in amounts of 1.5 to 2 kg per day, and of hydrazine hydrate, in amounts of 30 to 40 g per day, at the intake end of the make-up pumps. The isotope composition of the coolant was: 10^{-5} kg F^{18} per liter; radioactive inert gases (Xe^{133} and Xe^{135}) $2 \cdot 10^{-7}$; A^{41} to 10^{-5}; traces of Na^{24} (at transition regime up to $3 \cdot 10^{-5}$ kg/liter); traces of Mn^{56} (at transition regime up to $3 \cdot 10^{-5}$ kg/liter).

Starting on March 7, 1970, the second power unit commenced operating on power with boron control, at a boric acid concentration in the coolant ~2.0 g/liter. The water ratings remained practically unchanged, except for the pH of 7.5 to 10.0.

The presence of boric acid in the coolant alters some of the physical characteristics of the reactor core. For example, power generation in the core became ~15% more uniform. On the other hand, the use of coolant treated with boron in appreciable concentrations alters the reactor dynamics. Experiments carried out on the second power unit showed that the temperature coefficient of reactivity drops by a factor of almost 2.5 when boric acid is used (in a concentration of 2.9 g/kg): when pure water is used the temperature coefficient of reactivity was $4.3 \cdot 10^{-4} \deg^{-1}$ at the operating temperature, and in the presence of boron (concentrated stated above) the temperature coefficient of reactivity became $1.7 \cdot 10^{-4} \deg^{-1}$.

Reduction in the temperature effect of reactivity also lowers the degree of reactor self-regulation, but self-regulation of the reactor was retained to a sufficient degree in the presence of boric acid in concentrations up to 3 g/kg in the second power unit (when one turbogenerator is shut down with the whole unit on full power, the reactor parameters stabilize out through self-regulation of the reactor).

Protracted maintenance of the specified boron concentration in the primary-loop coolant, as well as the pattern of decrease and increase in boron concentration, were achieved during the process of adjusting the performance of the power unit with the boron coolant. This smoothed the way for maintaining the control rod assemblies in positions minimizing the unevenness factor of the heat release throughout the reactor core volume. The minimum heat release unevenness factor is attained when the 12th group of automatic control and reactivity compensation system was set in place at the medium position with respect to the core height (the reactor contains 12 automatic control and reactivity compensation groups, which are withdrawn, when the reactor is started up, in the order in which they are numbered). This makes it possible to operate the reactor over the entire campaign with a minimized heat release unevenness factor.

The use of steam pressurizers to compensate pressure changes in the primary-loop made it possible to do without the introduction of nitrogen into the pressurizers, and operation was simplified on the whole, as a result. When the dynamic characteristics of the steam pressurizers in the second power unit were taken, they revealed excellent compensation behavior. The primary-loop pressure during transients, such as when scramming operations are initiated, varied less than in the first power unit.

The rated primary-loop pressure of 105 kg/cm^2 was maintained at a temperature of 313°C in the pressurizers. Whenever the primary-loop pressure rose above rating, injections from the delivery end of the GTsN pumps into the pressurizers commenced automatically, and the pressure then fell because of cooling of the vapor phase. The temperature in the pressurizer (and consequently the primary-loop pressure as well) is maintained automatically by a temperature regulator which acts on the transformer supplying the working group of electric heaters. The autotransformer voltage can be varied over a range from 0 to 380 V.

The second power unit was brought up to 100% power output level (365 MW(e)) on April 14, 1970 (see Table 3).

Complete dumping of the load occurred only once during the startup period, because of some malfunctioning in the electrical supplies of the control rod system.

The startup and adjustment operations revealed some shortcomings in the flowsheet and in the process equipment, and these were eliminated before the power unit was brought up to 100% power level (specifically, flaws in the seals of the control rod jackets on the reactor cover, and in the heating elements of the pressurizers, were eliminated, the reliability of the control rod system electrical supplies was improved, and jitter in the readings of the data-processing and computer machinery was eliminated).

Part 2. Operating Characteristics

The second power unit of the Novo Voronezh nuclear power generating station, with a power rating of 365 MW(e), was hooked into the power grid of the European part of the USSR in late December, 1969. This power unit occupies a special place among the VVER* power reactor installations in operation or under construction. This special position stems from the fact that the reactor in this second power unit (a V-3M reactor) is on the one hand a further development of the engineering concepts underlying the reactor (a V-1 reactor) in the first power unit of the power station, and in that case can be regarded as a modernized redesigned variant of that reactor, and on the other hand that the core of the V-3M reactor differs slightly, in its engineering characteristics and design solutions, from the cores designed for reactors with a power output rating of 440 MW(e) now being designed or built (the V-440 reactors). In that sense, the V-3M reactor is the prototype of the second generation of VVER reactors.

The concept underlying the development of the second power unit of the station was one of developing a reactor whose power output level would be well above (~75% above) that of the first reactor power unit, on the basis of verified engineering solutions and equipment already accepted in production (while experience in the design and fabrication of the reactor and equipment was already available through the building and operation of the first reactor power unit of the Novo Voronezh nuclear power station, and the Rheinsberg nuclear power station in the German Democratic Republic). This increase in power had to be achieved in a core retaining the previous dimensions (2.5 × 3 m), by increasing the specific heat loading and with the duration of the operating period between partial refuelings (averaging three per campaign) at ~6500 effective hours.

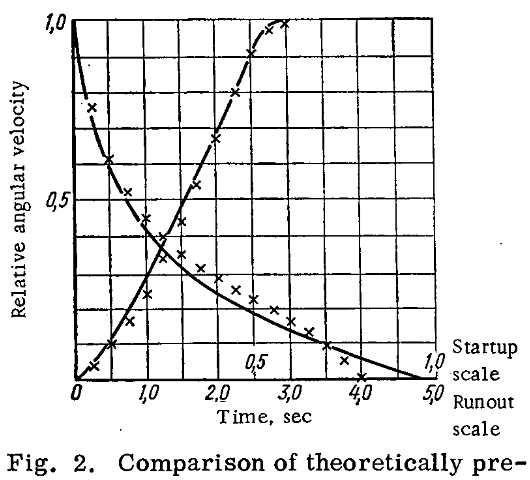

Fig. 2. Comparison of theoretically predicted pump runout curve and pump startup curve, with results of experiment run on reactor: ———) predicted curves; ×) experimental data points.

The possibility of increasing heat removal from the core was facilitated in two basic ways: by developing the heat transfer surface (through the use of 349 assemblies, each containing 126 rods 9.1 mm in diameter) and by bringing the mean values of the thermal parameters closer to the maximum values attainable by flattening out the power distribution field in the volume of the reactor. The power distribution field was flattened out by positioning the fuel, with enhanced multiplying properties, on the core periphery, and through the additional use of boric acid solution in the primary-loop

*VVER – water-cooled water-moderated power reactors.

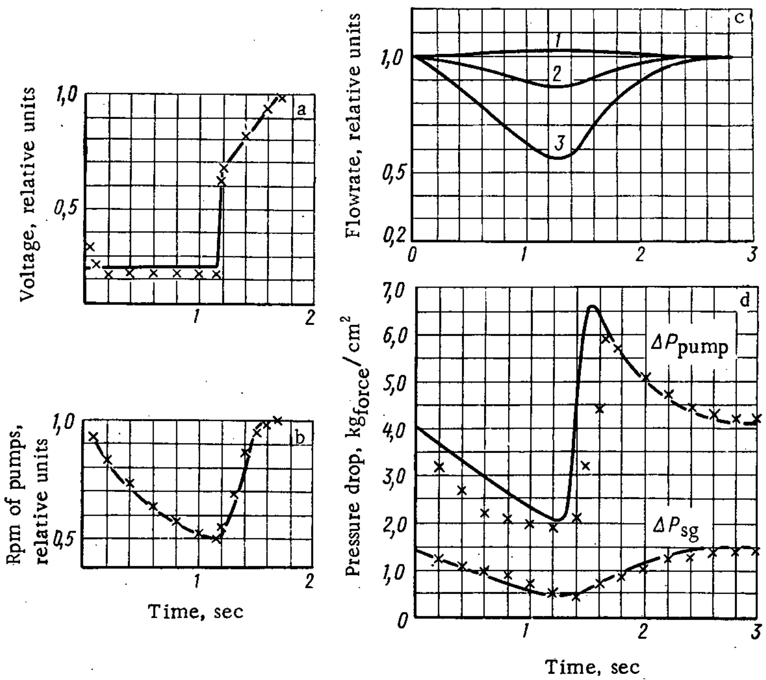

Fig. 3. Comparison of results from calculations of short-circuiting conditions, across 220 kV line, on two pumps with six pumps operating; with actual data (×) on changes in voltage (a), in pump rpm (b), in coolant flowrate (c), and in pressure drop (d) across pumps (ΔP_{pump}) and across steam generators (ΔP_{sg}): ———) theoretically predicted curves; 1) flowrate through subloops of four pumps; 2) flowrate through reactor; 3) flowrate through subloops of two pumps.

coolant, which minimizes the required number of absorbers of the reactivity compensation system partially inserted into the core. Reliance on liquid reactivity controls backing up a system of mechanical controllers (total of 73) capable of maintaining the subcriticality of the depoisoned reactor in the cold state with no boron present in the coolant makes it possible to draw upon industrial experience accumulated in the use of boric acid under conditions which were much less stringent from the standpoint of meeting nuclear safety requirements.

The stage-by-stage progress of the second reactor power unit to full power, begun in December, 1969, was completed on April 14, 1970 when the reactor was brought up to design parameters. The function of each distinct stage originated in the program of equipment testing and checkout, and bringing the reactor power unit up to power. Below, we discuss the checkout tests and their results, and their effect on regular operating conditions of the power unit and relation to the use of boron solution in the coolant.

In determining the allowable operating conditions for the reactor, the point of departure consists of the requirements for safe operation of the fuel elements under not only stationary conditions, but also expected nonstationary conditions and even emergency conditions. The power range is determined in line with the requirement that the thermal flux or the maximum increment in enthalpy does not reach values at which burnout on the fuel-element surface might occur. At the same time, there must be no severe meltdown of the central part of the fuel element, so that swelling of the fuel or disruption of the fuel-element jacket will not occur.

In VVER reactors using low-inertia pumps supplied with electric power from the base generators, in the primary-loop, possible deviations from proper core cooling conditions during operations stem mainly from breakdowns in the external electrical circuits of the power station. Even short-term interruptions (lasting up to 1 sec) in the electrical supplies will be serious in the case of low-inertia pumps, and will cut down their capacity considerably. Since uranium dioxide fuel is used, the process of extracting heat from

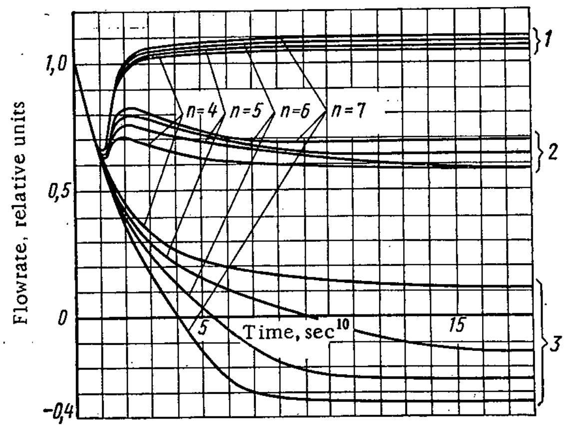

Fig. 4

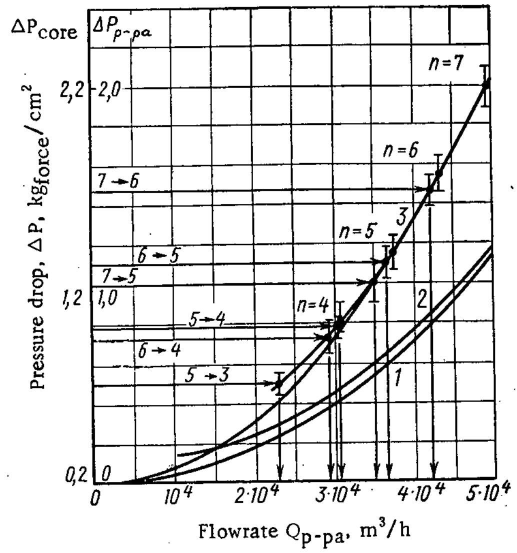

Fig. 5

Fig. 4. Changes in flowrate through reactor and subloops when all pumps are deenergized with short circuiting, over a time $\Delta\tau = 1.0$ sec, with subsequent restoration of rpm on all GTsN pumps except for two: 1) flowrate through subloops, with pumps resuming their previous rpm; 2) flowrate through reactor; 3) flowrate through subloops with pumps not yet recovering their previous rpm.

Fig. 5. Hydraulic characteristics of core at zero power, (1), of reactor core at rated power level (2), and of reactor at zero power (3): ——) theoretical prediction; ⫯) measured pressure drop across zero power reactor with n circulation subloops in operation and under conditions where pumps are shut off (7 → 6 with seven operating and one shut off); $P_{1k} = 105$ kg_{force}/cm^2; $T_{in} = 250°C$.

the fuel elements and transferring it to the coolant is a highly sluggish one, and the rate of change in reactor power, even in an emergency shutdown, is far below the emergency rate of reduction in coolant flowrate.

Those features which are decisive in the assignment of safety parameters for stationary operation of a VVER reactor are the nonstationary circulation conditions, in the light of the above. The behavior of these processes will be affected not only by the hydraulic characteristics of the circulation loop, but also by the dynamic characteristics of the principal circulation pumps (GTsN). A series of experiments was set up with the object of securing reliable information on the dynamic characteristics of the pumps while the startup of the second power unit was in progress, and these experiments were set up to conform with the scheme of the theoretical analysis of the process outlined below.

In order to describe the nonstationary circulation process, we made use of the following equations: $\tau_i(dG_i/dt) = \Delta P_i - \zeta_i G_i|G_i| - \Delta P_0$, the variation in coolant flowrate in the (i-th) circulation subloop; $\tau(G/dt) = \Delta P_0 - \zeta_0 G|G|$, the variation in coolant flowrate in the reactor. Here $\tau_i = \sum_j (L_j/S_j)(1/g)$ (L_j is the length of the loop interval with the open flow area S_j); $\tau = (L/S)(1/g)$ (L is the length of the common part of the loop with the effective open flow area S); $G = \Sigma G_i$; ζ_i, ζ_0 are the effective coefficients of hydraulic resistance of the subloops and their common part.

The head developed by the pump is represented, as a function of the rotation speed of the pump rotor ω_i and the flowrate G_i, in the form $\Delta P_i = A\omega_i^2 - B\omega_i G_i - CG_i|G_i|$. The rotation speed of the pump rotor is related to the pump dynamic characteristics by the equation

$$\mathcal{J}\frac{d\omega_i}{dt} = M_{\mathrm{rot}_i} - M_{c_i},$$

where $M_{\mathrm{rot}_i} = f(U_i, \omega_i)$ is the moment of the rotation and dependent upon the stress U_i and the rpm; $M_{ci} = f(\omega_i)$ is the resistance moment; $\mathcal{J}$ = const is the moment of inertia of the pump rotating parts. The

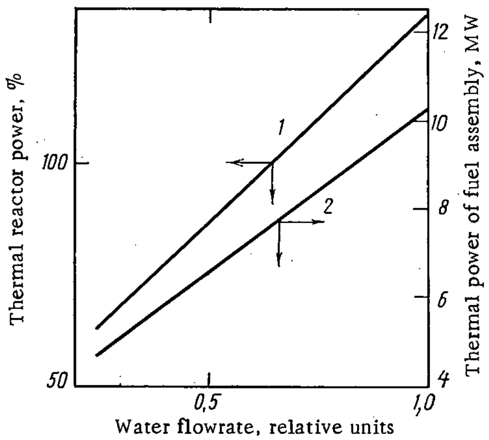

Fig. 6. Critical reactor power (1) and critical power output of fuel assembly (2) as functions of the relative flowrate of coolant through the reactor.

dependences $M_{rot_i} = f(U_i, \omega_i)$ and $M_{c_i} = f(\omega_i)$ can be found, consequently, by recording the pump runout (the pump runout is the change in rpm after the power driving the electric motor of the pump is shut off), and also the change in pump rpm during pump startup. The dependence $\Delta P_i = f(\omega_i, G_i)$ can be found if we record the time variation of the head developed by the pump, and the relationship between the head and the flowrate can be found if the time variation of the pressure drop across any interval of the circulation subloop is known. Oscillograms of the following parameters were taken during the tests with this purpose in mind: U = f(t) the supply voltage to the GTsN pumps; I = f(t) GTsN pump electric motor in-phase current; n = f(t) pump rotor rpm; $\Delta P_h = f(t)$ the pump head; $\Delta P_{sg} = f(t)$ the pressure drop across the steam generator of the circulation subloop with the pump tested.

Figure 2 shows a comparison of the predicted and measured GTsN pump runout curves and startup curves, confirming the possibility of making a theoretical description of the real process occurring when the pumps are started up and shut off.

The validity of extending the dynamic pump characteristics obtained in these experiments to other cases of nonstationary circulation, and the correctness of the description of changes occurring in the coolant flowrate through the primary loop are illustrated in Fig. 3, where results of theoretical prediction and experiment with three-phase short circuiting on a 220 kV line, lasting 1 sec, are plotted. Changes in the pump head and in the flowrate through the circulation subloop as functions of the pressure drop across the steam generator were recorded by low-inertia strain-gage pressure sensors.

The results obtained confirmed the correctness of the theoretical description of nonstationary circulation processes in the primary loop under probable operating circumstances brought about by outages of electrical supplies to the GTsN principal circulating pumps, and exerting a decisive influence on the level of allowable power when the reactor is operating in the stationary mode. The possible emergency conditions for lowering the voltage across all the pumps simultaneously with subsequent restoration of the circulation (by eliminating outages or malfunctions and operating the pumps with the turbogenerators coasting) on all GTsN pumps except for two,* are decisive in the choice of parameters for stationary operation of the second power unit.

Calculations of changes in flowrate under these conditions are illustrated in Fig. 4. Comparison of the theoretically predicted flowrate values at steady state and values obtained from the results of measurements of the pressure drop across the reactor can be made on the basis of the data plotted in Fig. 5. Comparison of curves 1 and 2 in Fig. 5 shows, in particular, the weak effect exerted on the hydraulic resistance of the core by the reactor power level when the typical power distribution among the fuel-element assemblies prevails. This state of affairs made it possible to make use of the constant value of the core hydraulic resistance coefficient. (The hydraulic characteristics of the core were plotted with the weight of a column of water 2.5 m high taken into account.)

As already pointed out, the initial reactor power was assigned such that burnout conditions on the surface of the fuel elements in any part of the core would be eliminated during the process of anticipated emergency reductions in coolant flowrate through the primary loop. The way the problem was solved is as follows. The values of the water flowrate through the fuel assemblies which would be attained in an emergency process are determined from the known changes in flowrate of coolant through the core, from the condition that the pressure drop across parallel channels be equal, and in line with the power-dependent hydraulic characteristics of the fuel assemblies. Further, on the basis of formulas characterizing burnout on the heat-transfer surfaces, a quantitative relationship between the power output of a fuel assembly and the minimum coolant flowrate through the fuel assembly at which the predicted burnout would occur on the surface of the fuel element under the conditions of heaviest heat loading of portions of the bundle of fuel elements in the assembly is arrived at.

*Here we consider the possibility of a malfunction in one of the generator switches remaining undetected for a long period.

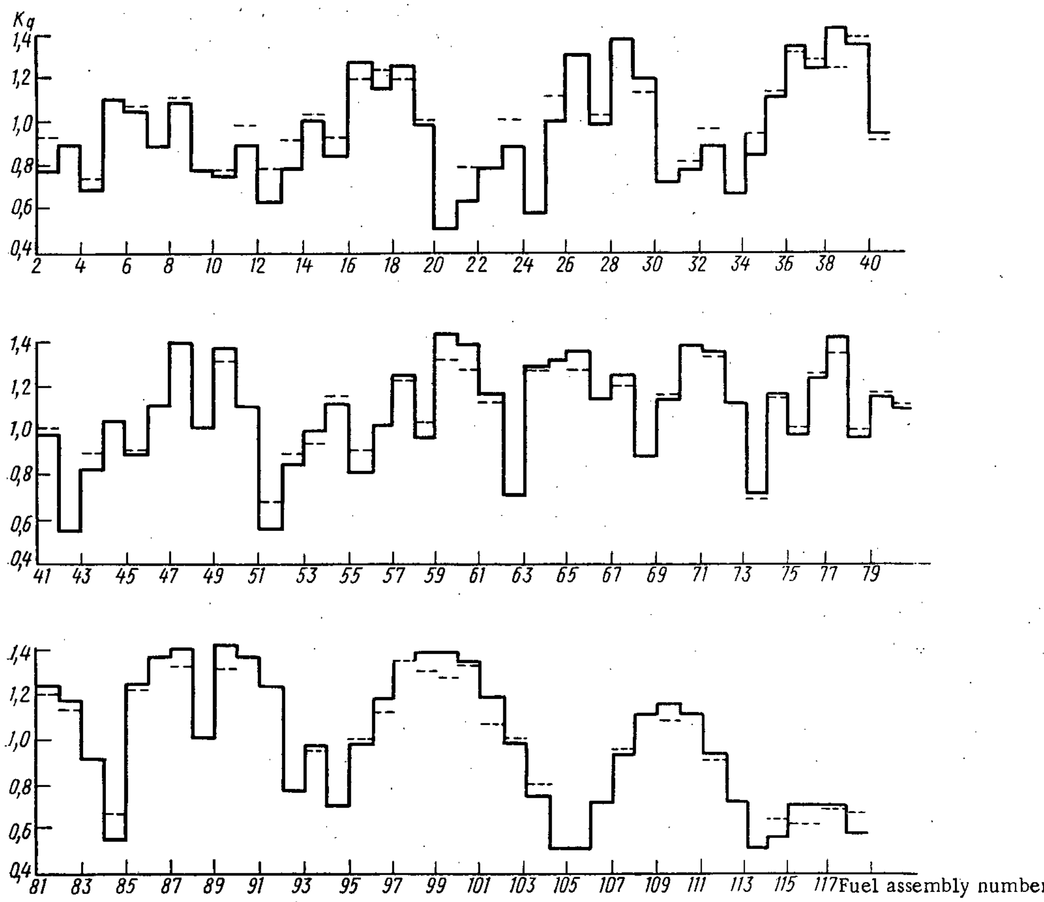

Fig. 7. Relative power distribution over fuel assemblies K_q of core of reactor in second power unit of Novo Voronezh nuclear power station, when rods of working group of control rod system are withdrawn 100 cm: ----) results of measurements averaged over symmetry sectors.

The orientation to minimization of the coolant flowrate is justified, in our opinion, by the following considerations. Viewing the problem of the nonstationary thermal conductivity of the fuel element in the simplified model (somewhat as was done by Tong [1]), presupposing instantaneous reduction in heat transfer at the initial instant and an equally instantaneous recovery within a certain time interval, we can arrive at an analytical solution of the time behavior of the fuel element cladding temperature. Such an analysis demonstrates that the elevated cladding temperature conditions resulting are retained for a fairly protracted time even after the initial rate of heat rejection to the coolant has been recovered. In this light, we can anticipate that recovery of the coolant flowrate under conditions where the cladding acquires an elevated temperature would not immediately eliminate the burnout beginning then, and the unfavorable process would be drawn out still further.

The dependence of the critical power of the fuel assembly (i.e., of the power at which burnout occurs on the surface of the fuel elements) on the coolant flowrate through the reactor, which we make use of in determining the allowable reactor operation conditions, is based on the results of experiments with bundles of rods of full scale length belonging to the reactor of the second power unit and is plotted in Fig. 6 (curve 2). At certain safety factor values and unevenness factors of the core power distribution between fuel assemblies, the data obtained are useful in determining the critical reactor power as a function of the coolant flowrate. This dependence is plotted in Fig. 6 (curve 1) for the case where the maximum power output of the fuel assembly exceeds the mean power output of the fuel assembly, averaged over the reactor, by 1.43 times. The relationship between the reactor power output and the relative water flowrate (relative to rating) through the reactor can be expressed by the formula $Q/Q_{cr} = 130\ (G/G_{rat})^{0.6}$ (in percentages of the critical or burnout power level). The (100%) power output rating corresponds to a flowrate through the reactor that is 64% of rating. This flowrate value is the minimum one in the mode where voltage across all the GTsN pumps of the primary loop is lowered for short time periods (see Fig. 4). The margin left till critical power levels are reached is 30% in the initial state.

This unevenness in the power output of the fuel assemblies is maximized in the case of rated reactor operating conditions where criticality is maintained by boron in the primary-loop coolant and by absorbers of only one control group which operates normally in the top half of the core. This value of the power distribution unevenness factor is in excellent agreement with predicted values, as attested to by the results, plotted in Fig. 7, of a comparison of theoretically predicted reactor power distribution between fuel assemblies and data obtained on the basis of recording the water temperature at the exit from the fuel assemblies. The relative power output of a fuel assembly was determined in this case from the formula

$$K_{qi}=\frac{t_{ik}^{ex}-\dfrac{\sum_j t_{j\,sl}^{in}G_j}{\sum_j G_j}}{\dfrac{\sum_i t_{ik}^{ex}g_i}{\sum g_i}-\dfrac{\sum_j t_{j\,sl}^{in}G_j}{\sum_j G_j}}\cdot\frac{g_i}{\dfrac{1}{m}\sum_{i=1}^{m}g_i},$$

where t_{ik}^{ex} is the water temperature at the exit from the fuel assembly in question; $t_{j\,sl}^{in}$, G_j are the water temperature at the point where the circulation subloop enters the reactor, and the rate of coolant flow through that subloop, respectively; g_i is the relative rate of water flow through the fuel assembly.

The use of boron to compensate reactivity excess contributed to flattening out the power distribution field in the reactor; this set of operating conditions is the fundamental one for the operation of the second power unit in the power station. At the same time, the presence of boron in the primary-loop coolant exerts an influence on such an important reactor property as the stability of on-power reactor performance.

The stability of on-power reactor performance is determined to an appreciable extent by the self-regulation behavior of the reactor, which is made possible by the negative power reactivity coefficient and negative coolant temperature reactivity coefficient. When boric acid is present, the temperature coefficient of reactivity drops below negative, i.e., the reactor becomes less stable against disturbances in the external load.

In order to check out the self-regulation capabilities of the reactor and the effect of boric acid on those capabilities, special tests were carried out with the disturbances in external load handled without bringing the reactivity compensation resources of the reactor into play. The experiments were conducted both with and without boric acid present in the primary loop, and as far as possible under identical conditions. The results obtained were also helpful in estimating the value of the temperature coefficient of reactivity and the effect of boron on that value. This becomes possible if we make use of the reactivity balance for the two states; before load disturbances were introduced and after stabilization of the transient process, i.e., $\alpha_H(\bar{t}_2-\bar{t}_1)+\alpha_U(Q_2-Q_1)=0$. When the power coefficient of reactivity $\alpha_U = 1.5\cdot10^{-4}\%^{-1}$, the average temperature coefficient of reactivity was found to be $-4.3\cdot10^{-4}\,\mathrm{deg}^{-1}$ for pure water at the working parameters, and $-1.7\cdot10^{-4}\,\mathrm{deg}^{-1}$ when the boric acid concentration in the water was 2.9 g/kg (at working stream parameters).

The values mentioned were obtained under the assumption that the dependences of $\partial\rho/\partial Q$ and $\partial\rho/\partial t$ on the power and on the temperature are linear over a small range of variation, and that the power coefficient of reactivity is independent of the boron concentration in the coolant. This approach is based on the fact that the changes in reactivity observed to accompany warming or cooling of boron-containing coolant are in excellent agreement with the value that might be brought about through changes in coolant pressure, and hence by changes in the volume content of boron in the multiplying lattice.

Even though the temperature coefficient of reactivity remained negative in the presence of boric acid, and the self-regulation capabilities of the reactor were retained, the sensitivity of the reactor to disturbances in the external load became enhanced when the boric acid content in the water was 2.9 g/kg. Assuming that the temperature coefficient of reactivity is a linear function of the boric acid content when the concentration of boric acid is 4.75 g/kg, we may expect a termination of negative feedback action with respect to the water temperature in the reactor of the second power unit (the boric acid content does not normally exceed 2.5 g/kg in operation at full power rating).

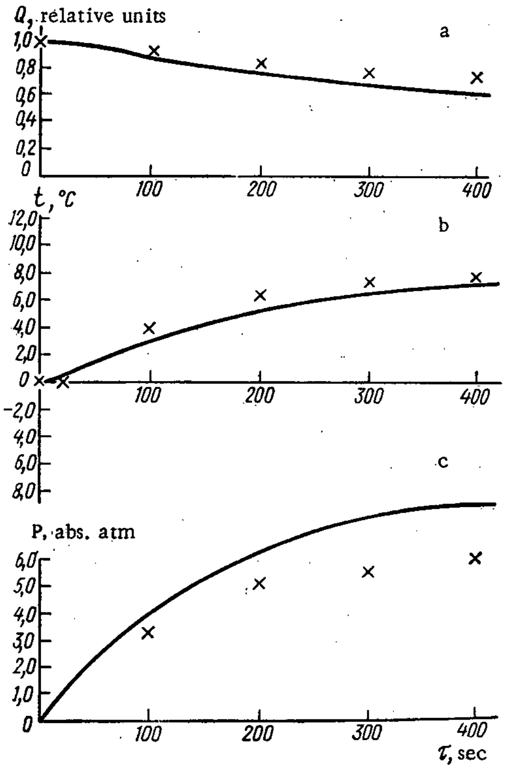

Fig. 8

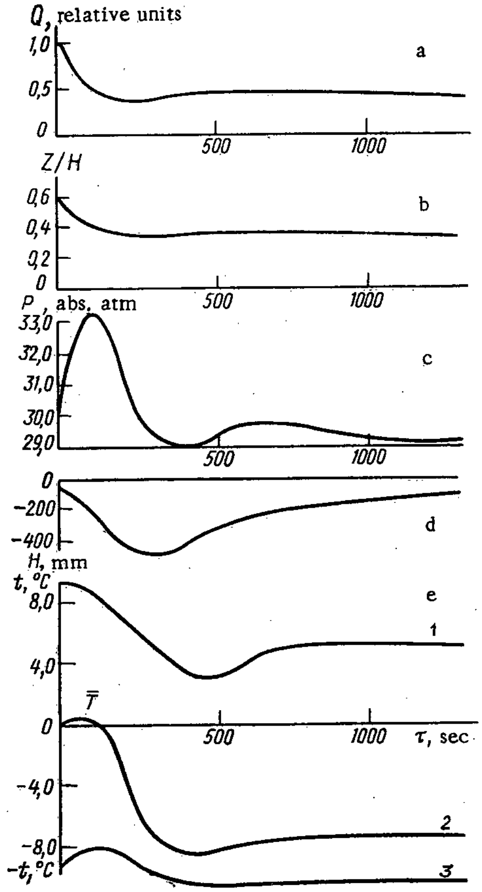

Fig. 9

Fig. 8. Changes in the basic parameters of the reactor in the second power unit of the Novo Voronezh nuclear power station, after dumping of load in self-regulation mode ($C_{H_3BO_3}$ = 2.9 g/kg): a) variations in reactor power output; b) deviations of mean primary-loop temperature from initial value; c) deviations in steam-generator pressure from initial value;———) theoretical prediction; ×) experimental.

Fig. 9. Variation in basic parameters of reactor in second power unit of Novo Voronezh nuclear power station when electrical load is dumped from 270 to 150 MW, and with automatic power controls operative: a) changes in reactor power; b) changes in position of working group of control rod system; c) changes in pressure in steam generator; d) changes in water level in pressurizer; e) deviations from initial value: of water temperature at entry to steam generator (1), of average water temperature in primary loop (2), of water temperature at entry to reactor (3).

The values of the temperature coefficient of reactivity obtained in these experiments are in satisfactory agreement with the results of direct measurements taken during the physical startup of the reactor [2], and with predicted values [3]. To illustrate the feasibility of a theoretical description of the processes under investigation, results of theoretical prediction and experiment on self-regulation of the reactor when the load is dumped are plotted in Fig. 8 (the predicted temperature coefficient of reactivity $\alpha_H = 2.0 \cdot 10^{-4} deg^{-1}$).

The data obtained in these experiments are useful in adjusting and testing the performance of the automatic power controller in response to a rapid dumping of the electrical power of the power unit from 270 to 150 MW (by 30% of the power rating). Disturbances are handled by the control group of absorber rods in such a way that the steam pressure in the steam generators is maintained constant over the specified range. The testing was done at a boric acid concentration of 1.46 g/kg in the primary-loop water, corresponding to the temperature coefficient of reactivity $-3 \cdot 10^{-4} deg^{-1}$.

As a result of the transient process (Fig. 9), absorber rods in the control group became displaced 46 cm, which corresponded to the predicted introduction of negative reactivity (0.0063). This value must compensate,

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clearly enough, the power and temperature effects of reactivity brought about by the 30% load change and by the 8% change in the average temperature of the primary-loop water, i.e.,

$$\Delta\rho = \Delta\rho_U + \Delta\rho_H = 1.5 \cdot 10^{-4} \cdot 30 + 3 \cdot 10^{-4} \cdot 8 = 0.0068,$$

which is in excellent agreement with the negative reactivity value introduced by the control elements. We can draw the following inferences from the above discussion:

1. Tests carried out on the second power unit of the Novo Voronezh nuclear power station (NVAES) provided the groundwork for arriving at safe operating conditions.

2. Comparison of theoretical predictions of some reactor parameters and their experimental counterparts proved the feasibility of the design and calculation procedures resorted to.

3. Introduction of boric acid into the coolant contributed to flattening out the power distribution in the core, while retaining the stability of reactor performance as a whole.

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EXPERIENCE IN THE OPERATION OF THE NUCLEAR POWER STATION AT RHEINSBERG

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UDC 621.311.25

The German Democratic Republic put its first nuclear power station into operation on May 9, 1966, at Rheinsberg [1]. It can be concluded from the four-year operation of the station that the operation as a whole is very satisfactory and that, compared with the projected goals, there exist certain reserves allowing an increase in the generation of electric energy. Ample experience with all technological aspects was gathered during the four-year period. More than 1.5 billion kWh had been produced by the end of the four-year operation. This figure could be reached because the operation was reliable and stable, as can be inferred from the station-load graph (Fig. 1) and from the total electric output graph (Fig. 2).

Good results were also obtained as far as the efficiency of fuel use is concerned. To date, four fuel recharges were made in the nuclear power station of Rheinsberg (Table 1). The unloading of a relatively small number of fuel elements at the end of the first operation period was necessitated by the transition from the initial operating conditions to conditions allowing a continuous operation in winter (transition made in 1968). The station was therefore recharged twice in 1968, and the second operation period of the reactor was relatively short. A comparison of the actual results with the projected goals shows that the actual fuel use was much better than the planned use. This conclusion is confirmed by the average burn-up of 11,750 MW · days/t, which exceeds the projected burn-up by 35%.

However, these data do not suffice for a full evaluation of the nuclear station's operation. The capacity of an electric power station is usually characterized by the coefficient of use in time and by the coefficient of fixed power output.

It follows from a comparison of the coefficient of fixed power output of the Rheinsberg Nuclear Power Station (Table 2) with the corresponding data of foreign nuclear power stations (in which pressurized water reactors with mean annual coefficients of fixed power output of 57.6% and 62.3% are typical), that the Rheinsberg Station rendered good results. The data listed in Table 2 prove that the Rheinsberg Nuclear Power Station can fulfill the requirements even in the winter period when the demand for electric energy reaches peak values. During workdays (including the daily peak periods of reactor load), the figures 97.3% and 94.4% were obtained for the coefficient of use in time and the coefficient of fixed power output, respectively, because the necessary short-time repair work was performed, in the majority of cases, at night or during off days. These figures prove that the optimum operation of the nuclear power station must take into account

TABLE 1. Fuel Recharging

Operating period	Operation period expressed in effective workdays (compared to 183 projected workdays)	Number of unloaded fuel elements (compared to a projected number of 44 fuel elements)	Burn-up of the unloaded fuel elements (MW · day/t)	
			average	maximum
1	302	27	5000	6400
2	177	37	8000	10,800
3	227	47	11,750*	15,700*
4	253	42		

* In the case of fuel elements which had been in the reactor from the beginning of its operation.

Translated from Atomnaya Énergiya, Vol. 30, No. 2, pp. 116-122, February, 1971.

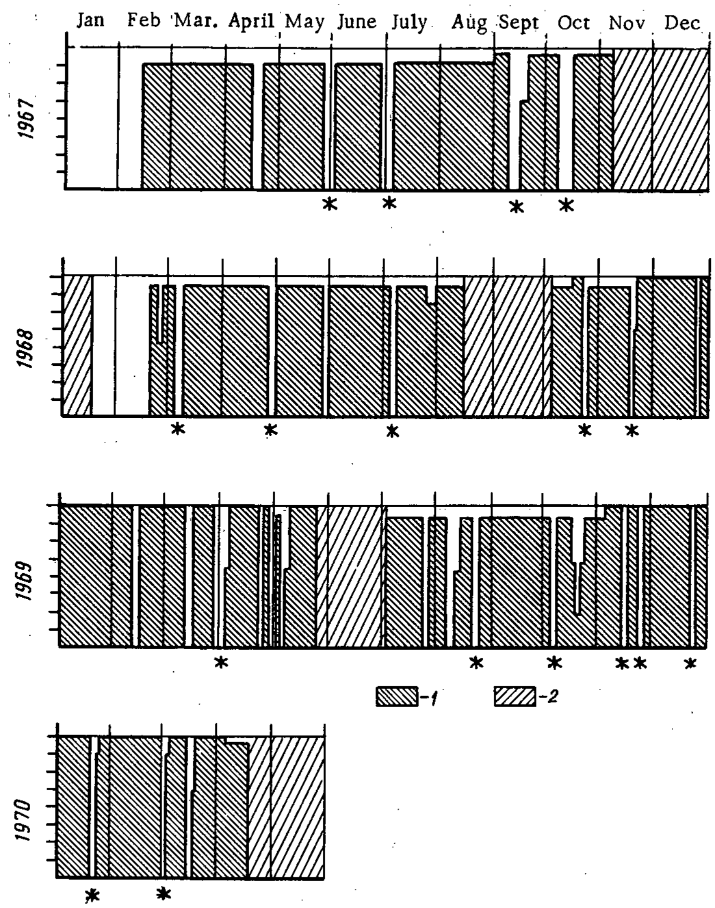

Fig. 1. Graphs representing the load of the nuclear power station: 1) normal operation; 2) fuel recharging; without shading – down periods resulting from breakdowns in the equipment; *) planned down periods.

periodic preventive maintenance work and the removal of irregularities which might disturb the normal exploitation of the station. In each year, about half of the total downtime consists of the fuel recharging period and the time spent in rechecking the reactor, both factors reducing the coefficient of use in time. A detailed analysis of the extent and the sequence of the work performed in the nuclear power station (including the fuel recharging work) proves that less than 25% of the total downtime is directly spent in recharging the fuel, beginning from checks of the hermetic enclosure of the fuel-element jackets and ending with the preparation of the reactor core for mounting the cover. The curves prove at the same time that the duration of a down period depends strongly upon the type of the work performed on the reactor, beginning from the removal of the cover and ending with the start-up of the reactor. The duration of a down period is additionally influenced by the time required for checking the equipment in the first circuit. Since the time periods between checks of the principal equipment are not always of equal length, the total time spent for this type of work is not the same in all years. The equipment revision work in the second circuit is usually performed within the planned time periods, provided that no unforeseen conditions develop, such as turbine repair work.

Even though the down periods can be reduced to some extent by well-planeed preliminary work, the down periods are, in principle, determined by the design and the type of equipment used. This statement is particularly valid for the reactor and the protective devices. In the development of future nuclear power stations, the designers must make provision for facilitating the repair work and must minimize the requirements of preventive maintenance work. Thus, the performance of a nuclear power station can be improved even in the planning stage.

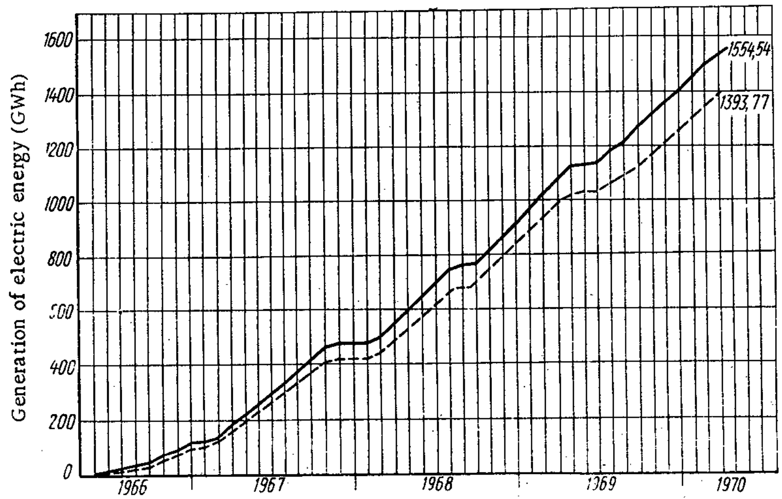

Fig. 2. Generation of electric energy in the nuclear power plant: ——) gross; ----) net.

The exploitation coefficients which were obtained in the Rheinsberg Nuclear Power Station prove convincingly that the principal equipment operates reliably. This can be said with regard to the operation circuits, as well as to the electrical measuring instruments. The positive overall evaluation is not lessened by the fact that breakdowns and disturbances, with partial cut-offs of certain equipment, occurred during the operation of the nuclear power station. The data of Table 3 show that in the nuclear power station of Rheinsberg (as in other nuclear power stations), the greatest number of breakdowns occurred in the second circuit, i.e., in that part of the nuclear power station on which less stringent quality requirements had been imposed during the time of the station's assembly. Breakdowns in the first circuit resulted, in part, from malfunctions of the electrical elements in the protection and control system, which caused a shut-off of the emergency-protection system. Failures in the water-supply system occurred in addition to disturbances such as damage to the coils of the fuel-element holding magnets of the emergency-protection system and to the housings of the actuating motors, which, however, did not exceed the anticipated down periods. Strong vibrations which interrupted the auxiliary pipes resulted from jolts produced by the piston pumps in the pipe system. An attenuator assembly reduced the vibrations, though they could not be completely eliminated.

Ruptures in the vanes of the high-pressure turbine cylinder constitute the principal problem in the second circuit. Though these ruptures were only a minor contribution to the number of breakdowns, their contribution to the total downtime of the nuclear power station is very important. The vane ruptures result from vibrations whose origin is currently under investigation. In view of the strong corrosion of the brass diaphragms, steel diaphragms were inserted, which resulted in a considerable improvement. Owing to special operating conditions and changes in the drainage system, no damage to the turbine vanes occurred in the last operating period. The operation conditions selected also had a favorable influence upon the operation of the diaphragms.

As far as the dosimetry and the radiation shielding are concerned, good results were obtained during the operation of the nuclear power station at Rheinsberg. No problems were encountered in obeying the

TABLE 2. Time- and Power-Coefficients of Reactor Exploitation

Year	Coefficient of use in time (%)	Coefficient of fixed power output (%)
1967	61.0	59.0
1968	67.7	66.1
1969	71.2	71.3
1968/1969 (water season)	91.6	92.9
1960/1970	86.4	86.7

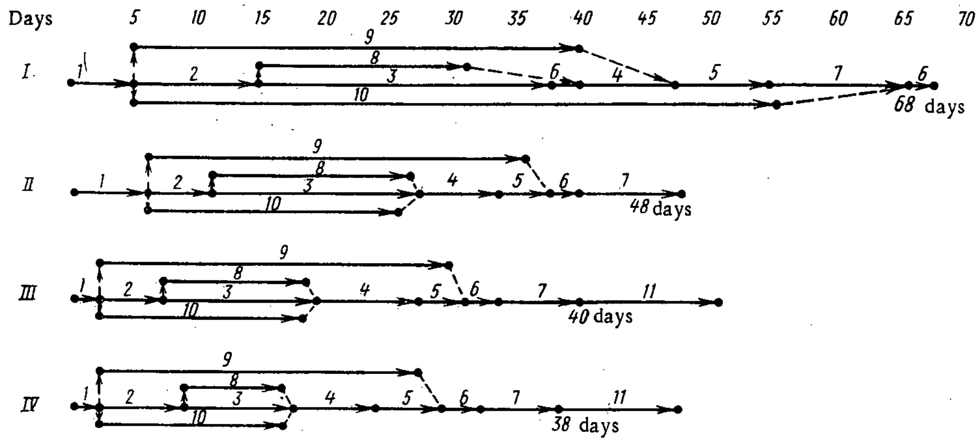

Fig. 3. Net representation of the recharging and checking operations (the Roman numerals denote the fuel recharging): 1) shut-down cooling of the reactor and deactivation of the first circuit; 2) preliminary work for, and removal of, the reactor cover; 3) inspection of the fuel elements and fuel recharging; 4) assembly of the lid and pressurization of the reactor; 5) insertion of the control and safety rod and checks of the cable strands; 6) physical tests; 7) tests of the shielding and of the performance of the equipment; 8) revision of the control and safety rod; 9) revision and checks of the measuring apparatus and instruments; 10) revision and x-raying of the first circuit; 11) measurements on the reactor.

admissible figures in the release of activity in the form of gases or aerosols into the air during the entire operation period. Tests proved that a close relation exists between the integral burn-up of the fuel elements and the concentration of the inert gases released during an operating cycle (Fig. 4). The maximum release of radioactive material at the end of an operating period and the release at the beginning of each operating period prove that a relation exists between the total burn-up and the total time during which a fuel element remained in the reactor. The leakage of gases from individual fuel elements distorts this relation to some extent. It was established that the leakage of gases from the fuel elements increases the release of radioactive material only in the first few days. In order to check this condition, one fuel element with a defective jacket was retained in the reactor during the fourth operating period. The performance of the fuel elements was excellent. This is witnessed by the fact that 37 fuel elements remained in the reactor during the four operating periods, i.e., they remained in the reactor one year longer than planned, but no defective jackets were found in these fuel elements. Seven fuel elements were retained in the reactor so that their durability might be checked during the fifth operating period.

Extensive measurements performed in the nuclear power station of Rheinsberg have shown that the degree of radiation shielding exceeds the projected values. The average radiation does not exceed 6% of the tolerable value [3]. The very low irradiation doses received by the service personnel of the station are a proof of the low radiation dose. Particular attention (involving some work) must be paid to the prevention of excess contamination of the premises before, during, and after repair work and checks. Interestingly enough, during the entire operating time of the Rheinsberg Nuclear Power Station, the irradiation of the operating and repairing personnel never exceeded the tolerable annual dose. Figure 5 shows the total external irradiation doses of the entire personnel for the various operation cycles of the reactor. It follows from Fig. 5 that, as could be anticipated, the personnel receive the maximum irradiation during recharging and checking work. In order to keep the annual irradiation dose below the tolerable levels, part of the personnel was temporarily removed from work which had to be performed under irradiation. In

TABLE 3. Number of Breakdowns in the Nuclear Power Station during 1967-1970

Point of breakdown	1967	1968	1969	1970 (to May)
First circuit	10	8	10	4
Second circuit	25	12	27	8
Electrical equipment	5	4	6	0

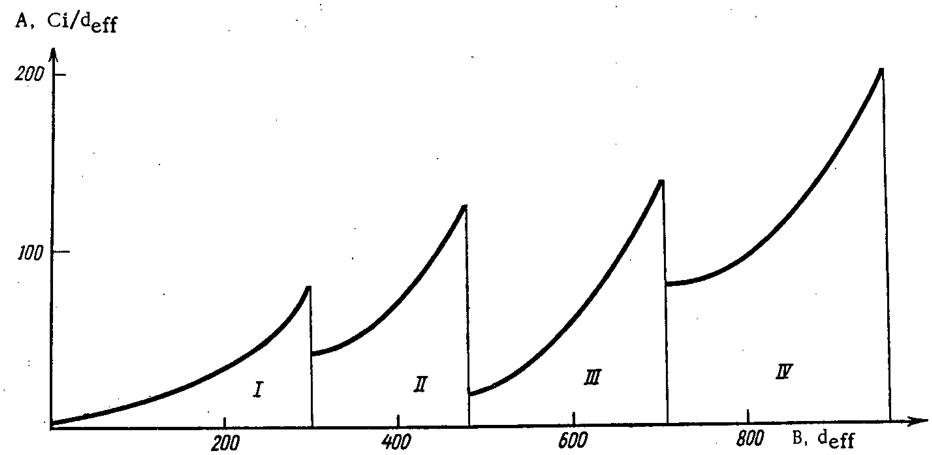

Fig. 4. Specific release of activity A (of inert gases) as a function of the burn-up B of the active reactor area during the first four operating periods (Roman numerals denote the operating periods).

planning work related to maintenance and the number of personnel required, one must take into account both the amount of maintenance work and the possible irradiation doses. This obviates the need for using equipment and designs which do not involve much repair and service work.

After the Rheinsberg Nuclear Power Station had been put into operation, extensive investigations were made for the purpose of establishing in accurate form the performance data when the station was put into operation, of becoming familiar with complex operations, and of providing for stable operation of the equipment. After that, the operational reserves of the principal equipment were determined for the purpose of establishing, with the aid of the data obtained, the greatest possible output of the nuclear power station. It was found that the output of the station could be increased to 80 MW (el.) without substantial changes and additional capital investments. However, this increase in the power output could not be obtained during the entire year, because in order to prevent a thermal overload of the fuel elements, the inhomogeneity coefficient of energy generation must not exceed 1.9, and the temperature of the cooling water must maintain a vacuum of 0.04 kg/cm^2 in the condenser of the turbine [4].

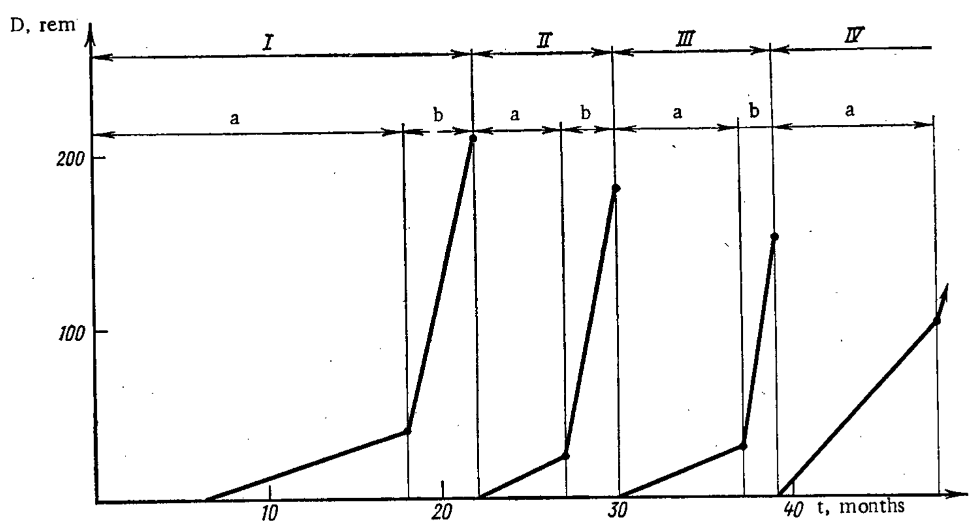

Fig. 5. Irradiation of the personnel of the nuclear power station during the first four operating periods (Roman numerals denote the operating periods): a) exploitation period; b) period of fuel recharging and checking.

Extensive chemical investigations were made in order to optimize the operating conditions. Particular attention was paid to the chemistry of the water. The chemical conditions of the water were constantly improved and, at the same time, methods for further improvement of the water treatment system were developed. The technique for deactivating the first circuit is of particular interest; the method had been developed in the nuclear power station of Rheinsberg and was used with great success on several occasions [5-7].

The tasks of the nuclear power station at Rheinsberg go beyond the safe operation of the nuclear power station and investigations concerned with the increase in the station's output. The goals are related to the construction and use of future huge power plants in the German Democratic Republic.

The scientific and technological developments call for the full use of the experimental possibilities for systematic investigations which can presently be made in stations with a fixed power of less than 100 MW. The results of the investigations made in the Rheinsberg Nuclear Power Station will be taken into consideration in new nuclear power stations. One must pay particular attention to the development and testing of techniques and methods for applying the available results to larger stations. Apart from this, one must bear in mind that newly developed equipment must be tested under real operating conditions. A method for regulating the power of a reactor by using boric acid and by treating the boron-containing heat-transferring medium of the first circuit is currently in the development stage in the nuclear power station of Rheinsberg. Investigations have been initiated with the goal of determining the burn-up of the fuel elements by experiments.

In order to obtain a high performance of new nuclear power stations, the experience gathered in the Rheinsberg Nuclear Power Station and the results of the investigations must be carefully evaluated and passed on to both designers and the service personnel of future power stations. To do this effectively will be possible only if an organized transfer of the results obtained is made. A special school was founded for this purpose in 1969 in Rheinsberg. Experienced specialists of the nuclear power station acquaint GDR engineers with the various problems of nuclear power generation. In the next few months, this school will teach the service personnel of the nuclear power plant "Nord," the second GDR nuclear power plant, which is currently in the construction stage. All engineers and qualified technicians of this power plant will receive special training in Rheinsberg. If necessary, the school in Rheinsberg can be used for teaching specialists of the member countries of the Soviet bloc.

It follows from what has been said above that the nuclear power plant in Rheinsberg has been successfully operated for four years. The equipment of the station was completely developed during this period, the power reserves were determined, and it therefore became possible to increase the projected output of the nuclear power station. Experience gathered in the operation is of great importance for planners and designers of new nuclear power stations and indicates in which direction major efforts should be made for obtaining an efficient use of future nuclear power stations. Of equal importance is the transfer of the experience gathered to the service personnel of future nuclear power stations.

The Rheinsberg Nuclear Power Station, which is the first station of its kind in the country, reliably generates electrical energy and, in addition, helps to solve an entire set of problems related to the development of nuclear power production. The nuclear power station at Rheinsberg contributes to the generation of electric energy on the basis of atomic energy.

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DISTRIBUTION OF DEPOSITS AND ACTIVITY ON THE SURFACE OF THE EQUIPMENT AND SERVICES IN THE VK-50 SINGLE-LOOP REACTOR

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UDC 621.039.553.3

Experimental data on sources of radioactive radiation, the distribution of activity, and the dynamics of its growth in the deposits formed in the technological loops of reactor power plants are of great practical

TABLE 1. Basic Characteristics of the Coolant Water Conditions

Period of plant operation	pH	Hardness, mg-eq/liter	Sodium-24, Ci/kg	Iron, mg/kg	Iron-59, Ci/kg	Zinc, mg/kg	Zinc-65, Ci/kg	Copper, mg/kg	Copper-64, Ci/kg	O_2, mg/kg
					Reactor water					
October, 1965 to April, 1966	7,5—9,5	20—145	$(0{,}8—4)\cdot 10^{-5}$	0,2—1,0	$2\cdot 10^{-9}$	0,05—0,2	$8\cdot 10^{-7}$	0,05—0,6	$(1—5)\cdot 10^{-5}$	0,03—2
September, 1966 to February, 1967	8,0—8,6	25—77	$1\cdot 10^{-5}$	0,05—0,7	$2{,}5\cdot 10^{-9}$	0,01—0,025	$2{,}5\cdot 10^{-6}$	0,05—0,1	$8\cdot 10^{-6}$	0,2
April, 1967 to April, 1968	6,1—8,4	15—21	$7\cdot 10^{-6}$	0,05—0,3	$3{,}5\cdot 10^{-9}$	0,005—0,02	$1\cdot 10^{-6}$	0,008—0,02	$4{,}9\cdot 10^{-6}$	0,2
June to October, 1968	6,1—6,3	15—21	$7\cdot 10^{-6}$	0,05—0,3	$3\cdot 10^{-9}$	0,005—0,02	$1{,}3\cdot 10^{-6}$	0,008—0,02	$4\cdot 10^{-6}$	0,2
February to July, 1969	6,5—8	10—30	$1\cdot 10^{-5}$	0,04—0,25	$4\cdot 10^{-9}$	0,005—0,01	$1{,}5\cdot 10^{-6}$	0,005—0,01	$2{,}3\cdot 10^{-6}$	0,1—0,15
December, 1969 to April, 1970	6,0—6,5	10—15	$8\cdot 10^{-6}$	0,03—0,06	$4\cdot 10^{-9}$	0,005—0,01	$1{,}5\cdot 10^{-6}$	0,005—0,01	—	0,1—0,15
					Feed water					
October, 1965 to April, 1966	6,6—8,2	5—60	$(0{,}2—8)\cdot 10^{-7}$	0,1—1,3	—	0,08—0,2	$7\cdot 10^{-8}$	0,02—0,1	$4\cdot 10^{-8}$	0,05—2,0
September, 1966 to February, 1967	6,2—7,4	3—34	$2\cdot 10^{-9}$	0,05—0,2	—	0,02—0,05	$2\cdot 10^{-8}$	0,02—0,05	$1\cdot 10^{-8}$	0,02—0,05
April, 1967 to April, 1968	6,0—6,3	3—17	$2\cdot 10^{-9}$	0,04—0,2	$4{,}5\cdot 10^{-10}$	0,01—0,02	$4\cdot 10^{-9}$	0,005—0,01	$1{,}5\cdot 10^{-9}$	0,02
June to October, 1968	6,0—6,3	3—15	$1{,}5\cdot 10^{-9}$	0,04—0,2	—	0,01—0,02	$3{,}4\cdot 10^{-9}$	0,005—0,01	—	0,02
February to July, 1969	6,0—6,7	3—25	$5\cdot 10^{-9}$	0,04—0,15	—	0,005—0,03	$5\cdot 10^{-9}$	0,005—0,007	—	0,01—0,02
December, 1969 to April, 1970	6,0—6,5	3—5	$5\cdot 10^{-10}$	0,03—0,05	$4{,}7\cdot 10^{-10}$	0,005—0,01	$2{,}5\cdot 10^{-9}$	0,005—0,007	—	0,01—0,02

Note: 1. The blanks mean that the activity of the isotope in question was not measured at that time. 2. The mean value of the activity is given.

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TABLE 2. Basic Physicochemical Characteristics of the Coolant under Different Operating Conditions of the Atomic Electric Power Plant

Characteristic	Conditions	Coolant: turbine condensate	feed water	reactor water
Value of pH	Starting	6,5—6,0	6,6—6,3	7,0—8,7
	Steady-state	6,0—0,2	6,2—0,2	6,0—7,0
	Emergency	5,9—6,4	6,2—6,6	7,0—9,2
Over-all hardness, μg-eq/kg	Starting	30—3	100—3	150—15
	Steady-state	3	3	7—15
	Emergency	5—50	5—100	20—50
Iron, μg/kg	Starting	300—50	100—50	600—100
	Steady-state	50	50	50—100
	Emergency	50—150	50—75	100—200
Zinc, μg/kg	Starting	200—20	200—20	150—20
	Steady-state	20	10	20
	Emergency	20—30	20—40	20—50

Note: The rated values of the physicochemical quantities were reached in three days during the starting periods.

interest. These data would make it possible to use theoretical models and calculation methods for predicting the radiation conditions in future atomic electric power plants and solve problems concerning the shielding, the structural materials, and the water-chemical conditions. The experience gained in the operation of the VK-50 single-loop reactor is of great importance in this respect.

The atomic electric power plant incorporating the VK-50 reactor constitutes a single-loop plant with a vessel-type boiling reactor. The structural materials include stainless and carbon steels, brass, a zirconium alloy, etc. The surface area of pearlite steel in contact with the coolant of the first loop is equal to about 4000 m^2 (approximately 39% of the total surface area), the surface area of stainless steel is 1750 m^2 (approximately 6%), the surface area of brass (without the technological condenser, is 9100 m^2 (approximately 52%), etc.

The present article provides the data obtained during the reactor operation from October, 1965 to April, 1970. During this time, the reactor operated at a thermal power level of 60-140 MW, with the exception of the last stage of operation (February-March, 1970), when the power level was raised to 160-185 MW.

The addition of hydrazine hydrate to the feed water, specified in the design for suppressing radiolysis and securing the standardized value pH > 7.0, caused an increase in the corrosion rate of brass. Therefore, neutral, compensation-free water-chemical conditions were used after December, 1965. The basic characteristics of these conditions are given in Tables 1 and 2. During the first stage of operation (October-December, 1965), the activity of the water and steam in the reactor was mainly caused by the activation of

TABLE 3. Distributed Composition of the Coolant with Respect to Iron Compounds (1966-1968)

Period of plant operation	Turbine condensate: ΣFe on the filter, mg/liter	%	ΣFe in the filtrate, mg/liter	%	Feed water: ΣFe on the filter, mg/liter	%	ΣFe in the filtrate, mg/liter	%	Reactor water: ΣFe on the filter, mg/liter	%	ΣFe in the filtrate, mg/liter	%
January to April, 1966	0,06—0,185	80—88	0,01—0,04	12—20	0,1—0,18	63—76	0,05—0,08	24—37				
September, 1966 to February, 1967	0,01—0,06	60—78	0,006—0,02	22—40	0,01—0,05	50—66	0,01—0,03	34—50	0,01—0,05	50—62	0,005—0,03	38—50
March to July, 1968	0,04—0,1	65—83	0,022	17—35	0,05—0,07	72—83	0,015—0,02	17—28	0,03—0,08	58—82	0,018—0,025	18—42
February to December, 1969	0,015—0,04	65—73	0,006—0,02	27—35	0,02—0,04	50—74	0,006—0,05	26—50	0,012—0,05	50—76	0,004—0,04	24—50

TABLE 4. Activities of Identified Radioisotopes in Deposits on the Equipment and Services Ci/cm^2

Sampling location	Duration of reactor operation, effective days	Zinc-65	Iron-59	Cobalt-60	Manganese-54	Zirconium-95	Cerium-144	Ruthenium-106	Barium-140 + Lanthanum-140	Strontium-89
Universal upper unit of reactor lid (UUU)	85	$4,3\cdot10^{-6}$	$1,3\cdot10^{-7}$	$1\cdot10^{-7}$						
	155	$2,3\cdot10^{-7}$	$1,3\cdot10^{-8}$	$2,8\cdot10^{-8}$						
	518	$1,5\cdot10^{-5}$	$1,1\cdot10^{-7}$	$2,6\cdot10^{-7}$	$3,1\cdot10^{-7}$	$3,2\cdot10^{-7}$	$1,5\cdot10^{-6}$	$1,7\cdot10^{-7}$		$8,7\cdot10^{-10}$
Frame of turbine LPC*	85	Traces								
	155	$3\cdot10^{-10}$	$<1\cdot10^{-11}$	$<1\cdot10^{-11}$						
	518	$9,7\cdot10^{-10}$	Not det.	$2\cdot10^{-10}$	$4\cdot10^{-12}$					
Brass tube of turbine condenser	85	$1,9\cdot10^{-9}$		Traces						
	155	$1\cdot10^{-9}$	$<8\cdot10^{-11}$	$<8\cdot10^{-11}$						
	518	$2\cdot10^{-10}$	—	$7,4\cdot10^{-11}$	$6,4\cdot10^{-11}$					
	780	$4\cdot10^{-9}$	$<1\cdot10^{-11}$	$3\cdot10^{-10}$	$3\cdot10^{-10}$					
Blade of the last stage of LPC*	85	$5,5\cdot10^{-11}$		Traces					Not det.	
	155	$1,5\cdot10^{-10}$	$<1,1\cdot10^{-11}$	$<1,1\cdot10^{-11}$					$1,0\cdot10^{-11}$	
	518	$4,3\cdot10^{-10}$		$9,4\cdot10^{-11}$	$6,4\cdot10^{-11}$					
	780	$2,4\cdot10^{-9}$	$2\cdot10^{-10}$	$2,2\cdot10^{-9}$	$9\cdot10^{-11}$					
Deaerator 1, bottom (deposit, Ci/g)	85	$2,9\cdot10^{-7}$	$2\cdot10^{-8}$	$4,3\cdot10^{-9}$						
	155	$1,8\cdot10^{-6}$	$1,4\cdot10^{-8}$	$2,4\cdot10^{-8}$	$2,4\cdot10^{-8}$				$3,2\cdot10^{-10}$	$4,3\cdot10^{-10}$
	518	$3,5\cdot10^{-5}$		$6,9\cdot10^{-7}$	$1,1\cdot10^{-6}$	$4,3\cdot10^{-6}$	$2,1\cdot10^{-6}$			
Deaerator 2, bottom (deposit, Ci/g)	85									
	155	$8,4\cdot10^{-7}$	$1,8\cdot10^{-8}$	$2,0\cdot10^{-8}$	$9,1\cdot10^{-9}$					$6,2\cdot10^{-10}$
	518	$1,2\cdot10^{-6}$		$8\cdot10^{-8}$	$1,8\cdot10^{-7}$					
Steam pipe beyond SND,† Ci/cm^2	518	$1,5\cdot10^{-7}$	$1,1\cdot10^{-8}$	$1,4\cdot10^{-8}$	$2,4\cdot10^{-9}$	$1,5\cdot10^{-8}$	$1,5\cdot10^{-10}$			

* LPC - Low-pressure cylinder.
† SND - Low-pressure system.

corrosion products (copper-64 and manganese-56), impurities (sodium-24), and coolant nuclei (nitrogen-13, nitrogen-16, and fluorine-18). After January, 1966, the activity of the iodines increased, this was due to a reduction in the gas-tightness of the fuel elements. Subsequently (from September, 1966 to April, 1968), fragment isotopes were identified in the coolant (cerium-144, cesium-137, barium-140, lanthanum-140, etc). After June, 1968, the activity of the fragment elements in the reactor water increased by two orders of magnitude. Gas activity peaks over 24-h periods reached 1500-2000 Ci. The activity of the dry residue increased; it amounted to $(1.2\text{-}4.8)\cdot10^{-4}$ Ci/kg in the reactor water, $(0.9\text{-}2.2)\cdot10^{-5}$ Ci/kg in the reactor steam, and $(0.6\text{-}1.4)\cdot10^{-6}$ Ci/kg in the turbine condenser. During this time, the activity of iodine-131 in the reactor water varied from $6\cdot10^{-7}$ to $6\cdot10^{-6}$ Ci/kg, the activity of iodine-133 and iodine-135 reached $1.7\cdot10^{-5}$ Ci/kg, the activity of molybdenum-99 reached $2.5\cdot10^{-6}$ Ci/kg, and the activity of barium-140 reached $6.2\cdot10^{-7}$ Ci/kg. This was due to a worsening of the defects in the leaky fuel elements transferred to the core periphery during the preceding reactor operating period. During reactor operating periods Nos. 3 and 4, these fuel elements were removed from the reactor, which reduced the fragment activity by a factor of more than 10. The activity of sodium-24 in the reactor water varied in accordance with the amount of inleakage of the technological water through the condenser and the reactor power level. The proportion of elements of corrosion origin in the coolant and the activity of these elements varied insignificantly in the course of service, with the exception of the first stages of reactor operation.

Since experience showed that iron compounds were the basic components in the deposits on the inside surfaces in the loop, we investigated the particle-size spectrum of these components along the water circuit (Table 3). The table indicates that the initial period of operation is characterized by the largest amount of insoluble suspensions in the coolant. Under steady-state conditions, ion-colloidal particles accounted for approximately one third of the total iron content in the coolant. Investigations have shown that the particle size is 1.2-3.5 μ for most of the suspensions (65-70%).

TABLE 5. Chemical Analysis of Deposits on Surfaces Inside the Reactor

Sampling date	Sampling location	Content of elements, g/m^2						
		Fe	Zn	Cu	Ni	Cr	Ca	Mg
1.3.1967	Peripheral mechanism of CSR*	0,63	0,104	0,002	—	—	0,046	0,092
11.3.1967	Lower part of the CSR* 09-35	1,47	0,91	0,125	—	—	0,106	0,022
"	Middle part of the CSR* 09-35	2,8	1,58	0,11	—	—		0,596
"	Upper part of the CSR* 09-35	4,3	1,23	0,13	—	—	0,7	0,144
16.3.1967	Areas between the steam openings of reactor	17,5	11,0	0,71	—	—	0,18	0,036
18.10.1968	Surface of the FECM† tube of UUV (smear)	0.71	0,33	0,24	0,004	Not det.	Not det.	Not det.
"	Inside surface of lubrication cups of UUU	0,51	0,05	0,29	0,003	Not det.	Not det.	Not det.
"	Outside surface of lubrication cups of UUU	1,01	0,35	0,125	0,023	0,002	Not det.	Not det.

* CSR – Control and safety rod.

† FECM – Fuel-element can monitoring system.

The methods of γ-spectrometry, radiometry, and chemical and radiochemical analysis were used in investigating the deposits and their activity along the loop of the plant. The sampling was performed according to the commonly used methods.

Visual inspection of the internal reactor fittings has shown that a friable hematite deposit predominates on the surfaces in contact with the steam (the top of the reactor, the shaft, etc.), while a dense magnetite deposit predominates on the surfaces in contact with the water (the control rods the tubes for checking the airtightness of the cans, etc.). Hardness salts settle on the high temperature surfaces in contact with the reactor water. The largest amounts of corrosion products accumulate in the deaerators.

In no case were there obvious indications of corrosion failure. The drainage piping system of the turbogenerator set, where the erosion wear reached 100% (airholes), were an exception. This made it necessary to replace the carbon-steel drainage pipes with stainless-steel pipes.

Analysis of the activity in the deposits has shown that it is due mainly to elements of corrosion origin (zinc-65, iron-59, and cobalt-60) and negligible amounts of fragment elements. The activities of deposits

TABLE 6. Chemical Composition of Deposits on the Inside Surfaces of the Equipment

Sampling location	Sampling date	Element content converted to oxides, % by weight								
		loss on ignition	Si_2O	Fe_2O_3	ZnO	CuO	MgO	NiO	CaO	Cr_2O_3	Mn_3O_4
Bottom part of deaerator 1	1.3.1967	1,3	15,2	66,9	0,71	0,29	0,52				
Bottom part of deaerator 2	"	7,0	5,4	84,5	0,8	0,8	0,33	0,055	—	0,001	
Bottom part of deaerator 1	24.2.1968	4,4	8,34	77,5	0,57	0,14	0,02	0,033	0,03	0,043	
Stiffening fins of deaerator 1	4.3.1967	3,37	3,42	88,3	1,5	1,43	0,64	0,029	—	0,003	
Side surface of deaerator 1	29.11.1968	1,54	7,9	86,4	1,16	0,72	Not det.	0,21	Not det.	0,12	
Throttle valve (end-face)	17.3.1967	0,29	0,58	94,4	0,49	0,015	Not det.	—	Not det.	—	
Flame of the turbine low-pressure cylinder (LPC)	9.3.1967	31,4	2,1	33	0,054	0,036	1,83	—	—	—	
Brass tube	10.3.1967	2,38	20,04	38	0,1	5,6	0,96	—	—	0,28	M_3O_4
Blade of turbine LPC	9.3.1967	64,9	2,6	14,6	2,88	0,25	Not det.	—	Not det.	—	
Steam pipe of condenser	1967	1	1	97,4	0,72	0,02	0,44	0,03	0,44	0,015	
Steam pipe of condenser	1968	1,2	0,71	98	1,02	0,12	0,31	0,01	0,31	0,008	
Condensate pipe beyond LPC	1968	2,20*	0,02	93,4	0,59	3,05	Not det.	0,03	Not det.	0,008	0,16
Steam pipe ahead of turbine	1968	0,19*	0,05	9,9	2,70	0,57	Not det.	0,05	Not det.	Not det.	0,08
Deposit ahead of SND throttle	1968	0,40*	0,47	96,00	0,90	0,32	Not det.	0,01	Not det.	0,001	0,12

*Measurements performed twice.

in the services and equipment of the plant are given in Table 4, while the chemical analysis results are given in Tables 5 and 6.

It should be mentioned that the analysis data given in Tables 4-6 do not reflect the true dynamic characteristics of the growth of activity in the deposits, since the equipment under consideration was partially deactivated and the deposits were removed during the design and preventive work.

Rough estimates show that the activity of zinc-65 in deposits in the turbogenerator set is 0.2 Ci after 155 effective days of reactor operation and 0.5 Ci after 518 effective days of operation of 5 and 7% of the total amount of zinc-65 that has passed through the turbine with the steam. Approximately 90% of this activity pertains to the pipes of the turbine condenser.

CONCLUSIONS

1. Compensation-free neutral water-chemical conditions do not produce harmful effects involving corrosion processes and hence a high level of contamination of the surfaces of the fittings and services with radioisotopes of corrosion origin. Their activity level after 780 effective days of operation does not present any difficulties when repairing and servicing the equipment of the atomic electric power plant.

2. Most of the corrosion product particles have a size of 1.2-3.5 μ. Ion-colloidal particles account for about one third of all the impurities. The use of a Powdex filter in the feed channel reduces considerably the entry of corrosion products into the reactor.

3. The major share of the activity of deposits belongs to isotopes of corrosion origin, of which zinc-65 accounts for approximately 80-90% of the total activity. In spite of the prolonged operation of the atomic electric power plant with increased radioactive gas waste (approximately 1500-2000 Ci/day), there has been no substantial contamination of the surfaces of pipes and equipment by fragment isotopes. The radiation level of the equipment and services of the steam condensation channel does not hinder repair work or lead to overexposure of the operating personnel.

SOME SAFETY PROBLEMS IN NUCLEAR POWER PLANTS WITH WATER-COOLED WATER-MODERATED POWER REACTORS

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The fuel core of water-cooled, water-moderated power reactors (WWPR) must operate throughout the entire fuel-burning cycle so as to prevent the appearance of inadmissible quantities of radioactive fission products in the coolant in steady-state operation as well as under transient and emergency conditions. To satisfy this requirement, which is equivalent to prevention of excessive heating of fuel element jackets or of the appearance of excessive stresses in them, the following restrictions are usually imposed:

with a sufficient margin of safety critical heat exchange is prevented from occurring when the coolant approaches the boiling point;

with a sufficient margin of safety the uranium dioxide is prevented from reaching its melting temperature;

with a sufficient margin of safety stresses in fuel element jackets are kept below their critical value;

no uncontrollable rise in reactivity is allowed to occur;

dangerous fluctuations of reactor power are kept within safe limits.

In connection with the above, the reactivity control characteristics, the nuclear characteristics, the thermal and hydraulic characteristics, as well as emergency situations are examined in this article.

Reactor Control and Safety System

Variations of the reactor core reactivity can be arbitrarily classified as fast and slow. The first takes place when the reactor power level is changed and in emergency situations. The latter occur as a result of fuel burn-out and xenon or samarium poisoning and also in case of reactor cooling.

Accordingly, the reactor control and safety system is divided into a reactor power control system, reactor shutoff system, and a system for compensating slow reactivity fluctuations.

The reactor power control system must usually provide means for compensating variations of reactivity taking place when the reactor power is changed, while the efficiency of control elements and the possible rate of rise of reactivity must ensure that no danger of core damage can occur.

According to the first requirement the efficiency of the reactor power control system must be adequate for compensating reactivity changes due to the Doppler effect when the reactor power is varied from zero to full rated power, for compensating fluctuations in average water temperature and density within this range, and for neutralizing reactivity variations due to the motion of control elements in the control region of the core. In addition, an excess of reactivity must be provided for compensating slight load fluctuations at the end of the burn-up cycle.

The second demand is usually satisfactorily met if the rate of rise of reactivity does not exceed the delayed-neutron fraction (at the end of burn-up cycle) in 20 sec. This is explained by the fact that in WWPR the Doppler-effect time constant is usually 3-5 sec so that at the above rates of rise of reactivity there is no danger of prompt-neutron runaway. It must be stressed that this is true as long as the control elements are prevented from moving upwards at speeds much higher than normal. If such a possibility is taken into

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TABLE 1. Efficiency Requirements of Reactor Control and Safety Systems

Requirement	Reactivity, %
Doppler effect	1.4-1.6
Variation of average water temperature	0-1.0
Variation of average water density	0-0.1
Control range	0.5
Operational margin	0.1
Overall efficiency required for control purposes	2.0-3.3
Power fluctuations	1.4-2.7
Subcriticality	1.0
Efficiency of most effective control element	1.0
Overall efficiency for shutoff	3.4-4.7
Fuel burn-up	8.0-10.0
Xenon and samarium poisoning	3.0
Cooling	4.0-5.0
Overload subcriticality	2.0-10.0
Overall efficiency needed to compensate slow variations of reactivity	17.0-28.0

account, the maximum efficiency of control elements must be limited to a safe value usually equal to 1-2 delayed-neutron fractions.

The reactor shutoff system must ensure that the core is subcritical after the reactor is shut down in a hot state even if the most effective control element is completely out of the core. Thus, the reactor shutoff system must be capable of compensating changes in reactivity when the reactor power is varied from zero to full rated power, of ensuring core subcriticality after the reactor is shut down in a hot state, and of reducing the system efficiency when the most effective control element is stuck in its extreme top position.

A comparison of the tasks of the reactor power control and reactor shutoff systems indicates that the requirements as to the compensation of reactivity fluctuations due to reactor power changes overlap partially. Some of the control elements can thus be common for both systems. This allows a reduction of the overall required system efficiency but inevitably leads to common control loops making it necessary to design the control and safety systems so that any damage to the control system elements does not affect the operation of the shutoff system.

The system for compensating slow changes in reactivity must provide an excess of reactivity for fuel burn-up and for xenon and samarium poisoning, and ensure that the core is subcritical at environment temperature and in the case of overloads.

Typical variations of reactivity in WWPR on whose basis the efficiency of control, and shutoff, and compensation systems are determined are listed in Table 1.

The design must take into account a 10-20% uncertainty in reactivity variations. The efficiency of reactivity control systems must thus be correspondingly increased.

On the basis of data in Table 1 one can find the required efficiency of the reactor control and safety system as well as of the subsystems of which it consists. If the control, shutoff, and compensation systems are completely separate, the overall efficiency can be as high as 27-36% (allowing for a 20% uncertainty). Such an efficiency is quite difficult to obtain with only mechanical control elements. If compensation of slow reactivity fluctuations is provided by a liquid absorber, the required overall efficiency of the control and shutoff systems is reduced to 6.5-9.6% (with 20% uncertainty allowed for). Such an efficiency can be obtained with the aid of mechanical control elements. If the control and shutoff systems use common control elements, the required efficiency drops to 4.8-6.3%.

Combined use of mechanical reactivity-control elements and of a liquid absorber makes it possible to compensate the excess of core reactivity in WWPR of any power and to ensure the satisfaction of all the above safety demands.

Physical Characteristics of the Core

In addition to their effect on the control element's efficiency, physical properties of the core affect such reactor safety characteristics as reactor stability in the face of spatial power fluctuations caused by redistribution of xenon in the core, the power distribution in the core, and finally the reactivity coefficients.

The reactor stability to xenon fluctuations is established by analyzing the stability of the transient spatial power distribution. The probability of xenon instability increases in large cores and in the case of strong power distribution compensation. If dangerous power fluctuations can occur, a system must be provided for monitoring the neutron flux distribution in the reactor so that reactor power fluctuations can be suppressed in time. It has been found that 1000-3000 MW (th) water-cooled, water-moderated power reactors are insensitive to xenon fluctuations and require no special control elements.

To ensure core safety in the course of reactor operation, the ratio of maximum to average power density should be kept below the established threshold. During the design one must analyze the power distribution in the core for all expected operating conditions as well as in the course of fuel burn-up and prove that the reactor design characteristics can be realized well within safety limits.

To analyze the reactor core behavior under steady-state, transient, and emergency conditions one must know the coefficients of reactivity. Since these coefficients vary in the course of fuel burn-up and depend on other operating conditions (e.g., power and temperature level, degree of xenon or samarium poisoning, liquid absorber concentration, position of mechanical control elements, etc.), it is necessary to establish the range of these coefficients.

Typical ranges of variation of reactivity coefficients in WWPR with liquid absorber lie within the following limits:

water temperature coefficient of reactivity, defined as the change of reactivity per one-degree change in water temperature, lies in the interval from $+0.5 \cdot 10^{-4}$ to $-6.0 \cdot 10^{-4}$ 1/°C;

water pressure coefficient of reactivity, defined as the change of reactivity per unit change in water pressure, lies in the interval from $+0.3 \cdot 10^{-5}$ to $+5.0 \cdot 10^{-5}$ 1/(kg/cm^2);

water density coefficient of reactivity, defined as the change of reactivity per unit change in water density, lies in the interval from $+0.2 \cdot 10^{-1}$ to $+3.0 \cdot 10^{-1}$ 1/(g/cm^3);

fuel temperature coefficient of reactivity (Doppler effect), defined as the change of reactivity per one-degree change in fuel temperature lies in the interval $-2.0 \cdot 10^{-5}$ to $-3.0 \cdot 10^{-5}$ 1/°C;

power coefficient of reactivity, defined as the change of reactivity per one-percent change of reactor power from nominal, lies in the interval $-1.0 \cdot 10^{-4}$ to $-2.0 \cdot 10^{-4}$ 1/% of power.

The stability and reliability of dynamic characteristics of nuclear power stations with water-cooled, water-moderated power reactors is in a large measure due to the fact that the Doppler-effect and power coefficients of reactivity are negative in such reactors.

Thermal and Hydraulic Characteristics

Nonuniformity Ratios. In the design of WWPR one usually uses the following definitions of ratios of nonuniformity of heat production:

fuel assembly ratio of nonuniformity, i.e., the ratio of maximum to average fuel-assembly power in the core;

axial power distribution ratio of nonuniformity, i.e., the ratio of maximum to average power density along the fuel assembly;

local ratio of nonuniformity of power distribution in fuel elements in the assembly, i.e., the ratio of maximum fuel element power to the average power in the assembly.

General ratios of nonuniformity of the local heat flux and water heating in fuel assemblies are defined as the ratio of maximum values to values averaged in the core.

Safety Factor. Each one of the general ratios of nonuniformity is the product of nuclear ratios of nonuniformity, that describe the distribution of heat sources (number of fission events) in the core, and safety factors that allow for deviations from the design conditions.

In determining the safety factors one must take into account the statistical combination of production tolerances in fuel density and enrichment, in the diameters of fuel pellets and elements, and in the relative position of fuel elements; the water distribution at the fuel assembly inlet and the mixing of water flow among the different channels in the assembly must also be considered.

In addition, thermal and hydraulic analysis of the core must consider redistribution of the water flow, reduction of its flow rate in the hot channel of the assembly (as a result of increased hydraulic resistance due to surface or bulk boiling), maximum water temperature and pressure deviation, as well as excursions of thermal power in the course of reactor operation as a result of errors and dead zone of instruments, control system errors, etc.

Maximum thermal power is governed by settings controlling power reduction and reactor shutoff and by errors due to drift and setting irreproducibility of neutron-flux monitoring devices.

Heat Exchange Crisis. The approach of boiling crisis is accompanied by a rise in surface heating temperature so that the conditions of occurence and the exact location of boiling crisis must be known. It should be remembered that since the analysis is made with empirical relationships their extrapolation to other parameter-variation intervals can lead to significant errors.

A comprehensive analysis of heat exchange crisis requires at least the following data: a relationship describing the conditions of heat exchange crisis, the range of channel parameters and geometry in which this relationship holds, and the statistical probability of deviation of experimental points from this relationship.

From the point of view of the onset of heat exchange crisis, the core operating conditions can be classified as follows:

rated conditions (reactor power, flow rate, pressure, and input heat content of coolant are within rated values);

worst-case steady-state conditions (high reactor power and input heat content of water, coolant pressure reduced by an amount determined by monitoring and control instrument errors);

maximum heat output conditions (maximum reactor power determined by settings and by errors in the operation of the emergency protection system);

constant load conditions (similar to rated conditions but with a smaller number of operating circulation loops);

operation with reduced coolant flow rate;

operation with reduced pressure in primary loop.

An analysis of heat exchange crisis makes it possible to find the dependence of critical power on critical flow rate for various primary loop pressures. This dependence is an important tool for determining the safety margin for heat exchange crisis under various operating conditions.

Typical dependence of critical fuel-assembly power on the critical coolant flow rate in WWPR is given below:

Flow rate, kg/sec	Power, MW (th)
5	6
10	10
15	13
20	15
25	17
30	19

The most important operating mode from the point of view of heat exchange crisis in WWPR is the operation under reduced coolant flow rate conditions. Typical statistical number of fuel elements reaching heat exchange crisis under conditions of reduced coolant flow rate is shown below:

Power, %	Flow rate, %	Number of fuel elements reaching crisis, %
100	100	0
100	50	$2 \cdot 10^{-4}$
100	40	$3 \cdot 10^{-1}$
100	30	12

Fuel and Jacket Temperature. The fuel temperature in WWPR is limited by the melting point of uranium dioxide, while the temperature of fuel jackets is limited by the rate of corrosion of zirconium alloys in hot water.

The power safety margin of melting temperature at the hottest spot in the core of WWPR is usually 20–50% of full rated power. The permissible temperature of zirconium alloys is about 350°C.

Emergency Situations

Emergency situations can be arbitrarily classified as emergencies associated with changes in reactivity and emergencies caused by mechanical failure of the equipment.

The reactor control and safety systems are designed to prevent damage of fuel element jackets under transient and emergency conditions. All the more probable failures give rise to signals that operate the protection system. To avoid false operation and to prevent loss of protection when some part of the safety system fails, the system usually has three channels, the coincidence of any two of three signals being necessary to trigger the safety system. The analysis of emergency situations usually assumes that the most effective control element is outside the reactor core.

To emergency situations associated with reactivity changes belong: start-up trouble, uncontrollable rise of reactivity as a result of extraction of the control element or drop in the concentration of liquid absorber, inflow of cold water into the core, ejection of the control element, drop of the control element, collapse of the core, overload trouble, etc.

The following are some of the most dangerous emergency situations.

Unforeseen Start-Up Troubles. The probability of an unforeseen rise of reactivity during reactor start-up is very low but can nevertheless occur as a result of failure of the reactor control system or of the control-element drive system.

The following measures are usually undertaken to prevent start-up emergencies:

reactivity is inserted at a predetermined and controlled rate;

control elements are combined into preselected groups that do not change throughout the burn-out cycle. Only one group operates at any given time so that no control element is able to move outside its group;

the reactor safety system is triggered by signals indicating uncontrolled upward motion of a control element, uncontrolled rate of rise of neutron flux or increasing neutron flux level, too high water pressure or temperature in the primary loop.

The negative fuel temperature coefficient of reactivity limits the power. Avoidance of start-up troubles by the above means prevents core damage.

Unforeseen Inflow of Cold Water into the WWPR Core. This can take place in two cases: operation of the emergency feed pump and connection of the cold circulation loop. The first case presents practically no danger as the flow rate of cold water is low and the water is collected in the mixing chamber above the core and enters the core only after flowing through the entire primary circuit.

To eliminate the second case and to prevent fuel element damage one usually takes the following measures:

the technique of connecting the cold circulation loop to the reactor is perfected in the course of start-up and adjustment operations;

blocking devices are added to prevent operation of the pump and of the loop valve when the temperature difference in the reactor and loop is excessively high;

to prevent rapid inflow of large amounts of cold water into the core the time the loop valve stays open is selected so that the connected loop has a chance to be heated by the reverse stream of water;

means are provided to operate the reactor safety system when the level or rate of rise of the neutron flux becomes excessive.

Prevention of "cold" emergencies by these means practically eliminates the possibility of reactor core damage.

Ejection of Control Element. Special devices are used to prevent ejection of the control element by the pressure drop which occurs when the drive mechanism cooling jacket is damaged; another way to prevent such emergencies is to limit the maximum efficiency of the control element. If liquid absorber is used operation with all mechanical control elements, except a single regulating group, extracted from the core is considered normal.

In WWPR the maximum efficiency of a power control element is usually 1-2 delayed-neutron fractions and the Doppler effect is capable of reliable prevention of power rise. Thus, together with the Doppler effect, the above measures provide reliable protection of the reactor core from damage.

Emergency Situations Caused by Mechanical Equipment Failure. These include the following situations: reduction of coolant flow rate through the core, loss of coolant from the primary circuit, total load cutoff, damage of main steam line, damaged steam generator tubing, failure of the spent-fuel cooling system, etc.

The first two situations are the most dangerous ones. The first can occur as a result of pump breakdown or from a failure in the pump power supply. The following measures are commonly undertaken to prevent such emergencies: the pump power supply circuits are made as reliable as possible (sectionalized power buses, use of special-purpose generators, etc.); the mechanical inertia of the pump rotor is increased; the reactor power is reduced as rapidly as possible to a new level corresponding to the lower flow rate level; safety means are provided in case of electrical power system failure and reduction of water flow rate. As a rule, these measures provide reliable protection against core damage.

The danger of the second kind of accident depends entirely on the scale of probable primary-circuit damage assumed in reactor design. At present there is no reliable quantitative definition of maximum possible equipment damage but it is qualitatively considered that the greater the assumed possible damage the lower the probability that it will actually occur. Irrespective of the assumed possible damage, great emphasis is put on prevention of large-scale equipment damage and on total elimination of radiation injury of the population and staff. The following measures are undertaken to achieve this:

the materials, manufacture, and testing of the primary-circuit equipment must meet much more stringent quality demands than in conventional power engineering;

all basic equipment of the primary circuit is placed in hermetically sealed locations designed to withstand increased pressures resulting from accidental loss of coolant and fitted with special cooling and pressure reducing devices as well as with cutoff devices that operate in case the allowable pressure is exceeded;

a rapid pressure drop in the primary circuit operates a fast-acting protection system and causes water to flow from the feed system into the primary circuit.

As an additional security measure a protective belt is provided around nuclear power stations.

A large body of experience in the design and operation of nuclear power plants with water-cooled, water-moderated power reactors has been accumulated in recent years. There is every reason to consider such plants no more dangerous than conventional electric power plants. Radiation injury to the population is practically impossible, and any presumable emergency situation in nuclear power stations with water-cooled, water-moderated power reactors cannot be of a catastrophic nature.

PHYSICS PROBLEMS AT THE BELOYARSK POWER STATION REACTORS

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The first two reactors at the Beloyarsk nuclear power station belong to the class of water-cooled, uranium – graphite-channel power reactors. The main feature of these reactors is nuclear superheating of steam which requires the existence of two types of fuel channels (FC) in the reactor: evaporative channels (EC), in which preheating and partial vaporization of water occurs, and superheat channels (SC), in which steam is superheated to ~520°C. The fuel elements in the fuel channels are of the tubular type with unidirectional cooling which prevents the entrance of fission fragments into the coolant loop in an accident. This arrangement ensures a very favorable radiation situation at the station itself and in the surrounding areas.

Equalization of Energy Deposition over the Reactor. One of the important problems in the operation of a power reactor is the provision of an equalized distribution of energy deposition over the reactor. In the reactors at the Beloyarsk station, this is achieved by physical shaping: for a fresh loading, by the appropriate arrangement of FC with different uranium enrichment and control rods; for transitional and steady-state regimes, by additionally shaping the fuel burnup in FC along a reactor radius.

A reactor loading consists of 998 FC, 732 of which are EC with uranium enrichment of 2 and 3% (EC-2, EC-3), and 266 of which are SC located in a ring (first unit) and in the center of the reactor (second unit) alternating with rows of EC-2; the EC-3 are arranged at the periphery of the core. The number and location of the EC-3 were determined on the basis of physical calculations of the reactor core aimed at ensuring equalized energy deposition over the reactor and achieving the maximum permissible burnup in FC for a given fuel composition. The core calculations were based on a two-group approximation using a special computer program. To carry out the calculations in accordance with the FC arrangement, the core was represented by four cylindrical regions with radii R_1 = 175 cm (234 FC), R_2 = 268 cm (324 FC), R_3 = 316 cm (220 FC), and R_4 = 358 cm (220 FC).

As shown by the calculations and by operating experience, for large uranium – graphite reactors with relatively small neutron leakage, the neutron flux distribution over the reactor is mainly determined by the multiplication properties of the reactor regions. The values of the neutron multiplication factors in the various regions, $K_{\infty 1}$ = 1.013, $K_{\infty 2}$ = 1.021, $K_{\infty 3}$ = 1.043, and $K_{\infty 4}$ = 1.045, make it possible to ensure equalized neutron distribution along a reactor radius with a variability factor K_V = 1.20–1.25. The simplified computational scheme was checked by more precise methods – for example, by two-dimensional calculations which demonstrated its complete validity. Of course, the method mentioned requires a smaller amount of machine time than the more complex computational methods.

A neutron multiplication factor which increased toward the periphery of the reactor is achieved by the installation of FC with 3% uranium enrichment (EC-3). FC reloading and uranium burnup in the various regions are chosen so that, at the end of an operating period (before fuel reloading), the $K_{\infty i}$ corresponded to assigned values which ensured the required equalization of energy deposition. During an operating period, maintenance of the $K_{\infty i}$ of the reactor regions is accomplished by the appropriate introduction of control rods into the core. Operating experience with the reactors at the Beloyarsk station showed that the control rods are a convenient instrument for shaping the energy deposition distribution over the reactor, especially with some reduction in their effectiveness (see below). The energy deposition equalization achieved in the reactors for $K_V \simeq 1.20$ is one of the best in the world.

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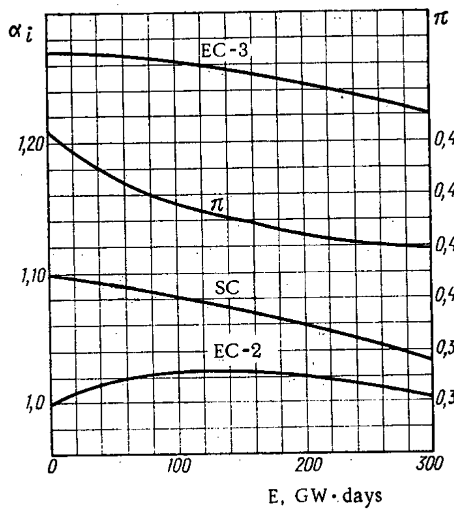

Fig. 1

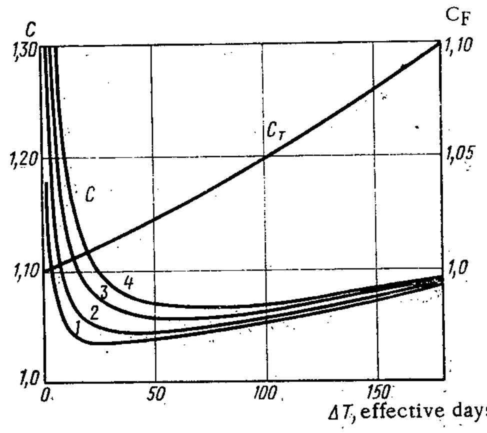

Fig. 2

Fig. 1. Dependence of the power ratio of FC (α_i) and of π on thermal energy (E) developed by the reactor of the second unit during the first operating period.

Fig. 2. Relative variation of total power cost (C) and of the fuel component of cost (C_F) as a function of time between FC reloadings (ΔT).

In the transitional and steady-state modes for partial FC reloadings, when the burnup excess reactivity is small ($\sim$1% for reactor operation in the period between reloadings), equalization of energy deposition distribution over the reactor is also accomplished by shaping fuel burnup in the FC. Reloading of FC of different kinds is carried out in such a way that the average neutron multiplication factor over the reactor regions corresponds to that given above.

Provision of Necessary Power Ratio in Superheat and Evaporative Loops. A characteristic of nuclear superheating of steam is the necessity of ensuring the required, and as constant as possible over the operating period, ratio of power in the superheat and evaporative loops ($\pi = N_{SC}/N_{EC}$). A value $\pi = 0.41$ permits one to obtain steam superheated to $\sim$520°C with optimal parameters for the thermodynamic cycle. The number of SC was chosen so as to ensure the aforementioned π during steady-state modes of partial reloadings with equalized energy deposition distribution over the reactor ($K_V \simeq 1.25$). The steady-state mode is characterized by a small variation in π over the period between reloadings (no more than 1% for constant neutron flux distribution along a reactor radius). As pointed out, the arrangement of SC in the first and second reactors at the Beloyarsk station is not the same: in the first reactor, they are arranged in a circular region bounded by radii of $\sim$1.75 and $\sim$3.1 m; in the second reactor, they are in a central region with a radius of $\sim$2.6 m. The ring arrangement of SC (first unit) has the advantage that the fraction of power going into superheat is only slightly sensitive to changes in radial neutron distribution; for the second unit with centrally located SC, the quantity π depends significantly on nonuniformity of neutron flux distribution along a reactor radius, and for neutron fluxes falling off toward the periphery, is characterized by the following data:

K_V	1,20	1,36	1,53	1,78
π	0,408	0,429	0,452	0,494

However, the central arrangement of SC (second unit) is preferable because it enables one to obtain a somewhat larger value of π (by $\sim$12%) for the same number of SC. In addition, the central arrangement of SC, which assure better multiplying properties than with EC, makes it possible to increase neutron utilization in the reactor, increasing the average fuel burnup by $\sim$10%.

The demand for stability of π carries with it the requirement for maintaining an equalized distribution of energy deposition along a radius of the reactor core during an operating period. The steam temperature at an SC exit is a good means for monitoring the distribution of energy deposition over the reactor.

As pointed out, the number of SC was selected so as to ensure the required π in the steady-state mode, when π changes little. In the initial period of reactor operation with a fresh loading, the different burning

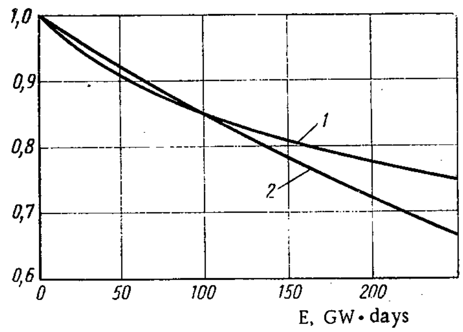

Fig. 3

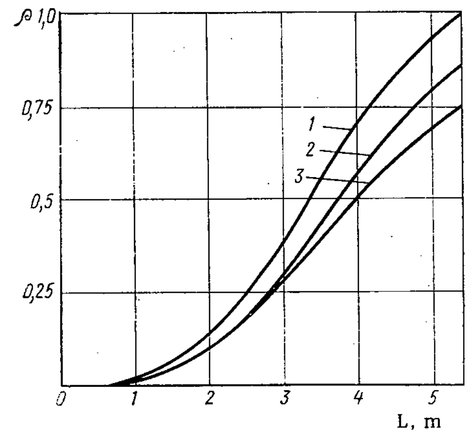

Fig. 4

Fig. 3. Relative variation in effectiveness (1) and boron content (2) as a function of thermal energy developed by a reactor.

Fig. 4. Relative effectiveness ρ of automatic control rod as a function of displacement for E = 230 GW•days.

rates of EC and SC leads to an imbalance of loop power. Figure 1 shows the dependence of the change in power of FC of various types (the power of a fresh FC was normalized to the power of an EC-2) and of the quantity π on the thermal energy devloped by the reactor. The calculation assumed that the reactor operated with $K_V \approx 1.25$. The power in the superheat loop falls particularly rapidly with respect to that in the evaporative loop at the beginning of operation; this is explained by the relatively small change in EC power because of the somewhat greater breeding of fuel in the slightly enriched EC-2. In practice, the constancy of π was achieved by gradually loading SC to the planned number.

Use of a Combined Control System. As the initial excess reactivity is exhausted, when the number of control rods in the core is reduced to zero, the equalization problem with respect to energy deposition becomes particularly complex. The control rods (CR) presently used in the Beloyarsk station reactors (boron steel bushings 39 × 3 mm in size with 2% of boron by weight [1]) are comparatively heavy. The following chart shows the effect of a CR inserted into the core on the power of surrounding FC:

0,92	0,86	0,86	0,92
0,90	0,74 CR	0,74	0,90
0,92	0,86	0,86	0,92

The significant "drop" in FC power on the introduction of a CR makes the energy deposition variability factor worse by an average of 5-7%. The relatively high effectiveness of regular CR places a limitation on the number of rods which can be kept in an intermediate position to avoid significant deformation of the neutron flux over the height of the core. The lack of "freedom" in selecting CR position makes the equalization of energy deposition along a reactor radius difficult. This was confirmed by operational experience when it was impossible to achieve the required equalization of energy deposition by means of the control rods remaining in the reactor at the end of the first operating period and the variability factor went from $K_V = 1.20$ to $K_V = 1.27$. A forced reduction in reactor power followed as a result. Because of this, it has presently been decided to switch to a combination control system.

The rods which were intended for compensation of the slowly varying excess reactivity in burnup and equalization of energy deposition fields, will be replaced by lightened "gray" absorbers made of ordinary heat-resistant steel with an effectiveness which is 2.5 times less than that of the standard CR. The standard CR will be retained to compensate for temperature effects and xenon poisoning. The use of lightened CR makes it possible to have a satisfactorily "soft" effect on the shape of the neutron field, to avoid large "drops" in power in FC adjacent to the rods, and to reduce the variability factor for energy deposition in the reactor. Calculations show that the use of lightened CR assures more stable distribution of energy deposition throughout the reactor during operation. The root-mean-square variation of EC power during periods between reloadings is ~4% while this quantity is 10% when standard rods are used. One of the advantages of the new, heat-resistant steel rods, in addition to those already noted, is the considerably lower heat deposition by radiation (about a factor of four less) which reduces the temperature in the rods themselves and in the metal structural parts; in contrast to standard rods the new rods are less subject to radiation swelling and do not change their effectiveness during irradiation.

Partial Reloading. After the end of the first operating period, it was expedient to carry out the replacement of burned out FC with fresh ones in small steps, moving gradually to the equilibrium state of the reactor. Reloading of FC in small steps makes it possible to increase significantly the depth of burnup in unloaded FC without increasing the U^{235} loading in the reactor and thus to reduce the fuel component of the cost of electric power. As has been shown [2], with an established scheme of partial reloading in a reactor consisting of one kind of channel and having a linear dependence of K_∞ on fuel burnup, the depth of burnup in spent FC is $B = 2B_0/1 + \eta$, where B_0 is the depth of burnup in FC for the first operating period and η is the fraction of FC reloaded per reloading.

One of the features of channel reactors is the capability for relatively simple FC reloading, including continuous reloading during reactor operation.

As already noted, the equalization of energy deposition distribution in established partial fuel reloading mode is achieved by shaping the burnup in the FC loaded. The presence of FC of different kinds (and different cost) in each region makes it possible to assure the required $K_{\infty i}$ of the reactor regions in many ways. It is obvious that there exists a ratio of FC of different kinds which corresponds to the minimum fuel component of electric power cost (C_F) while assuring the required equalization of energy deposition distribution over the reactor. Optimizing calculations were carried out using a special program which made it possible to determine the fraction of reloaded channels with minimum C_F for different intervals between FC reloadings. The energy deposition distribution with $K_V \approx 1.25$ assumed in the calculations provided shaping of fuel burnup in the channels loaded. The fuel component of electric power cost for various intervals between reloadings was obtained specifically for a reactor of the second unit at the Beloyarsk station with three types of FC (EC-2, EC-3, SC) including the actual nature of the relation $K_\infty = f(B)$ (Fig. 2). The quantity C_F was normalized to the value of the fuel component of electric power cost for continuous FC loading.

The continuous reloading mode makes it possible to increase FC burnup by ~80% in comparison with the first operating run of the reactor.

For reactors in which the replacement of FC requires shutdown and the associated additional expenses for down time of the reactor, there exists an optimum time interval between reloadings (or an optimum fraction of reloaded FC) which corresponds to a minimum value of the total cost C of electric power. Figure 2 shows a series of curves for the total cost of power as a function of the time between FC reloadings and of the down time for the reactor. The down time of a reactor comprises the shutdown time, cooling time, FC loading time, and time to get up to power. For the reactor down times considered, the optimum interval between FC reloadings falls within the limits of 30-70 effective days. The reloading time for a single FC was assumed to be 2 h. The total time for shutdown, cooling and startup was varied: 1 day (curve 1), 2 days (curve 2), 4 days (curve 3), and 6 days (curve 4). The thermal power of the reactor was assumed to be $N_T = 530$ MW. The total cost C of electric power was normalized to the value of C for the continuous loading mode without reduction in reactor power.

Purging Superheat Channels. A specific feature of the reactors at the Beloyarsk station is the startup mode [3]. The reactor is brought up to power without an outside source of heat. In accordance with regulations, startup is performed by the so-called SC purging process, which is associated with a relatively rapid replacement of SC water by steam. A reduction of reactor power from 10% to 2% of nominal with simultaneous reduction in feed water flow immediately precedes SC purging. In this situation, water begins to boil in the separator and SC purging occurs. The purging process is continued for about ten minutes. The change in core composition brought about by SC purging leads, on the one hand, to a change in the multiplying properties of the medium (particularly the shift in reactivity) and, on the other, to a change in the effectiveness of the inserted control rods.

The replacement of water by steam in the SC increases the effectiveness of the combined control system rods because of the increase in migration length and neutron leakage. The effectiveness of inserted control rods rises by ~5% on the average. Strictly, the shift in reactivity can be both positive and negative. The water in FC affects the multiplying properties of the lattice in two ways: it reduces the thermal neutron absorption, but it increases the neutron resonance capture escape factor. The sign of the effect depends mainly on the ratio of the amounts of moderator and uranium.

The 200 mm lattice spacing used in the reactors at the Beloyarsk station is only slightly sensitive to the presence of water in FC because of the small shift to the left from the physical optimum. The shift in reactivity is negative and varies from 0.2% for fresh regions to 0.4% for burnup regions. The variation of the shift in reactivity with burnup is explained by the relative reduction in the "weight" of the water in the slag region. As shown by reactor operating experience, SC purging is easily handled by the control units (on the average, by withdrawing four CR), and presents no difficulty for the operators.

Variation in Efficiency of Control Rods during Reactor Operation. With extended operation of a reactor having a control system of absorbing rods made of boron-containing materials (boron steel for the Beloyarsk station reactors), it appears necessary to consider the variation in the effectiveness of the controls because of significant burnup of the B^{10} in them. The change in rod effectiveness also results from a change in the diffusion properties of the material during burnup.

In a computer calculation of rod burnup, an equivalent cell of a regular lattice of rods was considered in the center of which a CR surrounded by a multiplying medium was located. The properties of the medium were made homogeneous with appropriate cross sections which took into account the heterogeneous arrangement of fuel in the reactor and its burnup. Figure 3 shows the computed results for the variation of the total effectiveness of a CR under conditions of continuous irradiation as a function of the thermal energy developed by the reactor (curve 1). The curve was normalized to the effectiveness of an unirradiated rod in a fresh region. Also shown is the relative variation of B^{10} concentration because of burnup (curve 2).

It is of interest to know the variation of concentration, and the associated distortion of the calibration curve for an automatic control rod, resulting from nonuniform burnup of boron over the height of the core. As a rule, automatic control rods are kept half-inserted into the core (for operation in the linear portion of the calibration curve), which increases the nonuniformity of boron burnup even more.

Figure 4 shows the nature of the variation in calibration curves because of nonuniform burnup of B^{10} along the length of a rod in comparison with the calibration curve for an unirradiated rod (curve 1). Curves 2 and 3 are typical of calculated calibration curves for rods which are continuously in the core and which are respectively inserted halfway and completely for ~230 GW·days thermal energy developed by the reactor (approximately the end of the first operating period for the second unit).

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START-UP AND OPERATION OF CHANNEL-TYPE URANIUM – GRAPHITE REACTOR WITH TUBULAR FUEL ELEMENTS AND NUCLEAR STEAM SUPERHEATING

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Two uranium – graphite channel-type reactors with tubular fuel elements and superheated steam parameters of 80-90 abs. atm and 510-520°C at the turbine inlet operate at the I. V. Kurchatov Beloyarsk Nuclear Power Plant in the USSR.

Nuclear power plants using nuclear superheating of steam and tubular fuel elements have a number of advantages: high efficiency, use of commerical thermomechanical equipment, and the possibility they afford of building nuclear power plants in populated areas by virtue of their high radiation safety.

For these reasons, the mastering of nuclear power plants with nuclear steam superheating by the industry is of considerable importance for the advance of nuclear power engineering [1].

The first Beloyarsk reactor with an electric capacity of 100 MW in conjunction with a VK-100-6 turbine has been put into operation (on April 26, 1964) [2]. The second reactor, started on December 29, 1967, has an electric capacity of 200 MW and operates in conjunction with two VK-100-6 turbines [3].

Apart from their capacity, the reactors of the first and second units differ in their flow diagrams and in the location of the evaporation and steam superheating channels in the core.

The first unit has a two-circuit flow diagram. The primary, boiling circuit includes 730 evaporation channels, separators, a steam generator, and the main circulation pumps. The secondary circuit contains 268 U-shaped steam superheating channels in which steam is heated to 520-540°C, a steam generator, and the turbogenerator with its auxiliary equipment. The steam superheating channels are located in the circular region and alternate with the evaporation channels.

The coolant flow diagram is shown in Fig. 1a.

The total number of fuel channels in the second unit and their construction is the same as in the first unit. The steam superheating channels are located in the central region of the core and also alternate with the evaporation channels. The flow diagram is a single-circuit one (see Fig. 1b).

Bubblers are provided in both units for heat removal during start-up.

The design of the Beloyarsk plant was based on theoretical and experimental investigations on electrically heated test stands and in the loop circuits of the First Nuclear Power Plant reactor whose purpose was to select and work out the technique and conditions of independent start-up [4]. In working out the start-up conditions we have considered problems associated with heating of the nuclear plant equipment, the change from water cooling of fuel channels to boiling in the evaporation channels and superheating in the steam superheating channels, and the additional rise of steam parameters and the unit capacity.

At the first stages of tests the transition from water cooling of the steam superheating channels to steam cooling (one of the basic steps of the start-up process) consisted in gradually replacing water by a steam-and-water mixture and then by steam at about 30% of full rated power. An experimental analysis of

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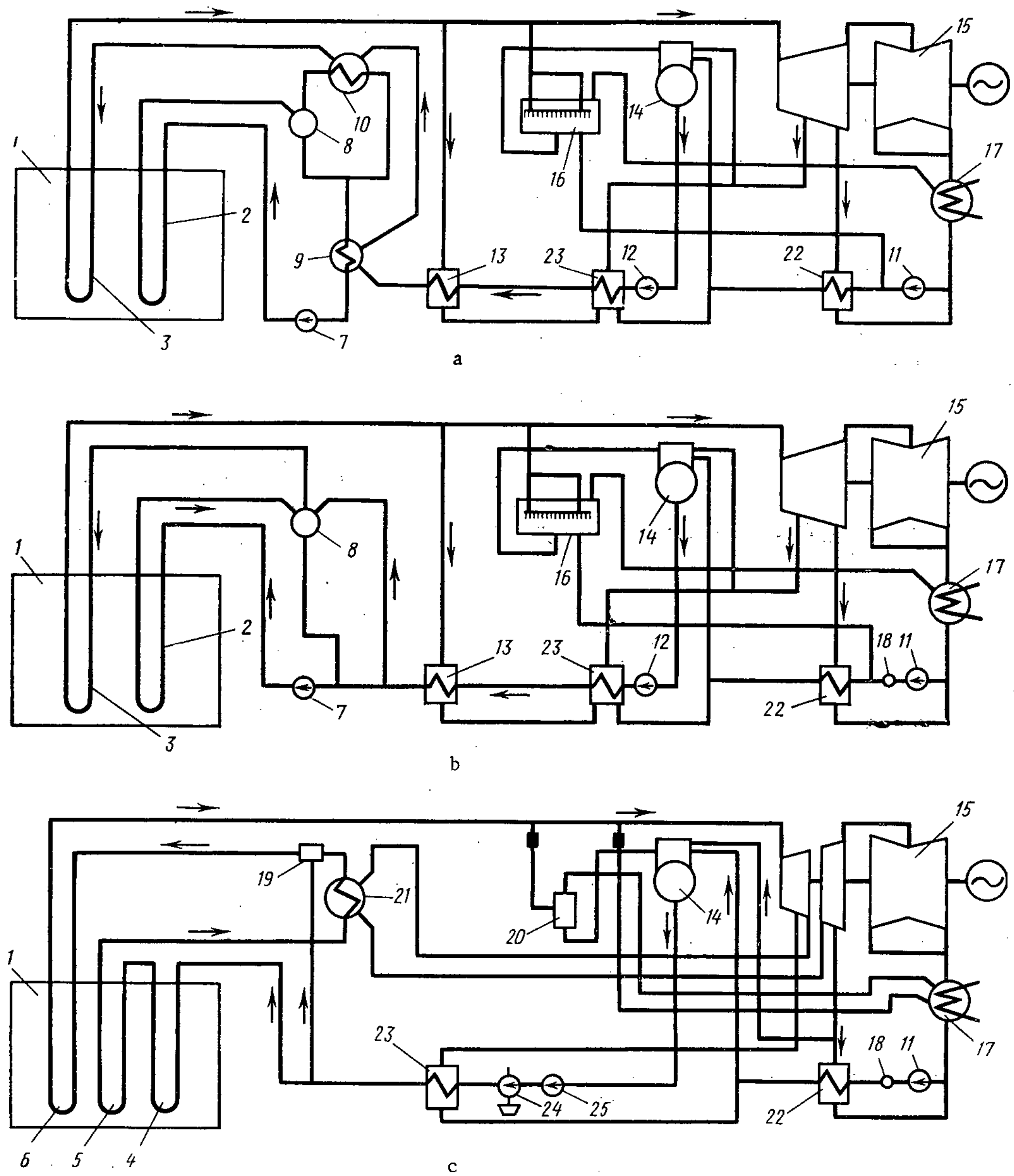

Fig. 1. Heat flow diagram: a) first Beloyarsk unit; b) second Beloyarsk unit; c) nuclear plant with supercritical coolant parameters; 1) reactor; 2) evaporation channel; 3) steam superheating channel; 4) preheating channel; 5) first superheating channel; 6) second superheating channel; 7) circulation pump; 8) steam separator; 9) preheater; 10) evaporator; 11) condensate pump; 12) feed pump; 13) superheat regulator; 14) deaerator; 15) turbogenerator; 16) bubbler; 17) condenser; 18) condenser purifier; 19) mixer; 20) start-up separator; 21) intermediate steam superheater; 22) low-pressure regenerative preheater; 23) high-pressure regenerative preheater; 24) feed turbopump; 25) booster pump.

this method in the First Nuclear Power Plant loop and on the test stand showed the presence of coolant-flow-rate pulsation, repeated water splashes into "blown channel," and in some instances also pulsation of fuel element temperature.

After calculations and experimental trials we have chosen the method of transferring the steam superheating channels to steam cooling at a reduced (to 2-3%) reactor power making use of heat accumulated in the coolant and in the equipment (mainly in the graphite stacking) that is released at reduced pressure in the steam circuit. In the course of development of this method we have found the minimum pressure drop between the collectors of saturated and superheated steam required for purging the steam superheating channel of water; the channel flow rates that provide hydrodynamic stability of the coolant at different powers and circuit pressures; and the hydrodynamic characteristics of the fuel channel at various pressures.

Physical experiments on the first-unit reactor in the course of start-up indicated that when water is removed from the fuel channel the reactivity effect amounts to $\Delta K/K = +0.07\%$ for 268 steam superheating channels and $\Delta K/K = -0.6\%$ for 730 evaporation channels.

During power start-up we have determined the basic purging parameters of the steam superheating channels such as the coolant flow rate and pressure, checked out the reactor power, tested the monitoring system performance and hydrodynamic characteristics of the circuits, and adjusted the flow rates in individual channels [5]. To check out the individual start-up stages (formation of levels in bubblers, separators, and evaporators; purging of secondary-circuit steam lines, flow rate distribution in the evaporation channels under boiling conditions, equipment performance, adjustment of automatic controllers), the reactor of the first unit was put into operation in late January 1964 with the evaporation channels only (726 channels). The steam superheating channels were fitted with devices simulating hydraulic resistance of the channels. The obtained results confirmed the design studies [6].

At the present time the units are started in the following order. For starting the first unit, the circuits are filled with deaerated water, water circulation is established, air is removed, the pressures in the first and second circuits are raised to about 100 and 30 abs. atm respectively.

The equipment is heated up at about 10-14% of reactor capacity. The average heat-up rate (according to water temperature in the separators) is maintained at about 30°C/h on the basis of experience gained with operation of barrel-type boilers, even if the reactor equipment allows considerably higher heat-up rates.

No heat removal is provided as long as the coolant temperature at the reactor outlet does not exceed 160°C. At 160°C a water level forms in the bubbler after which the excess heat is removed into the turbine condenser. The heat-up ends when the water temperature at the steam superheating channel outlet reaches about 230°C. The overall heat-up time is about 9 h.

The next step is to purge the steam superheating channels of water. The transient process in the secondary circuit takes place while constant pressure is maintained in the primary circuit and the evaporator channels are cooled without boiling. For purging the steam superheating channels the reactor power is rapidly reduced to $\sim 2\%$ of nominal and the flow rate of feed water is reduced so that a level forms in the evaporators. The steam-and-water mixture from evaporators and steam from the steam channel are led into the bubbler and then to the deaerator and turbine condenser.

The steam superheating channels are purged after a level forms in the evaporators. The purging process is monitored on the basis of the pressure drop between the distribution and outlet headers and the coolant temperature at the outlet of each steam superheating channel. To speed up purging of all steam superheating channels, gate valves in front of the bubbler are partially opened for 1-2 min and the pressure in the circuit is made to drop faster, this results in an additional discharge of steam. The rate of pressure drop is chosen to conform with the allowed temperature conditions of the steam generator and is about 1.5 atm/min. The time of level formation in the evaporators is 8-10 min, and that of steam superheating channel purging 6-10 min. After purging, the pressure and temperature of steam superheating are raised by partially opening the gate valves in front of the bubbler and by increasing the reactor power.

Two hours after the steam superheating channels are purged and when the reactor operates at about 10% capacity, the steam lines and turbine are heated and the turbine is prepared to be connected to the power line. Further increase of power takes place with the turbine connected to the line.

When the reactor power reaches $\sim 35\%$ and the steam pressure ~ 60 abs. atm, the primary circuit is transferred to boiling conditions and separator levels are formed. The operating conditions of the main circulating pumps are continuously monitored during the transient process, and a temperature margin of at least 5-6°C before boiling is maintained at the intake pipes of the main circulating pumps. Level formation in the separators is accompanied by a gradual change of pressure. The actual rates of pressure change are slightly lower than the calculated ones. The time till the controlled level is reached in the separators is about 3 h and depends only on the throughput of separator bleed lines. The specific features of a single-circuit

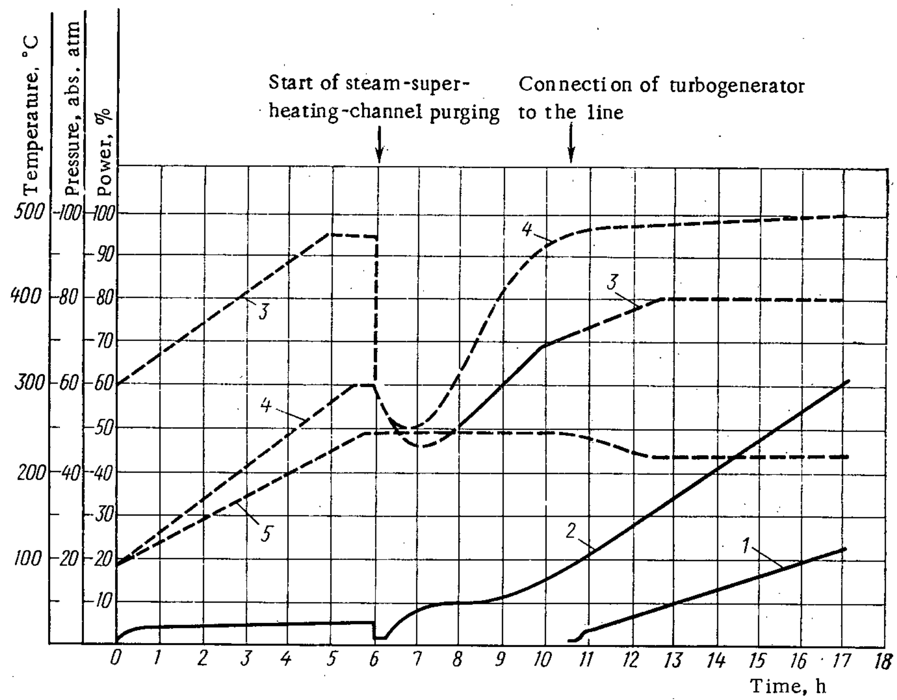

Fig. 2. Variation of basic parameters in the course of start-up of the second unit of the Beloyarsk Nuclear Power Plant: 1) turbogenerator power; 2) reactor power; 3) steam-line pressure; 4) coolant (water or steam) temperature at the steam superheating channel outlet; 5) feed-water temperature.

diagram make the sequence of start-up operations of the second unit somewhat different. Purging of the steam superheating channels and transition to boiling conditions in the evaporation channels take place simultaneously.

Filling of the circuits and equipment heat-up is the same as in the first unit. The final heat-up parameters are higher and amount to ~90 abs. atm and ~290°C. Two main circulation pumps are used to drive the coolant in the evaporation section of the circuit.

After heat-up the reactor power is reduced to 2-3% of nominal and purging of the steam superheating channels and transition to boiling conditions in the evaporation channels take place.

To form levels in the separators, the flow rate of feed water is considerably reduced, water is purged out of the separators, and the flow rate to bubblers is increased. As a result the water in separators and fuel channels boils causing the steam superheating channels to be purged of water and steam-and-water mixture. The purging operation is monitored as in the first unit. Both the main circulation pumps and the separator levels are monitored.

After purging of the steam superheating channels is completed, the reactor power is increased and steam flow into the bubbler is reduced while the rate of temperature rise of superheated steam is kept at about 1°C/min and the pressure drop between the steam headers at least 0.5-0.6 kg/cm^2.

The automatic level control system is put into operation as soon as the water in separators reaches the rated level.

The subsequent increase of reactor power, turbine preparation, and connection of the turbine to the line are the same as in the first unit.

Variation of the basic parameters in the start-up of the second unit is shown in Fig. 2.

In April 1964, 918 evaporation channels including 192 channels designed for superheated steam operation at 400-420°C were loaded into the reactor of the first unit. Because of reduced temperature, the steam pressure was ~0.1 abs. atm in the turbine condenser and 70 abs. atm at the input to the turbine.

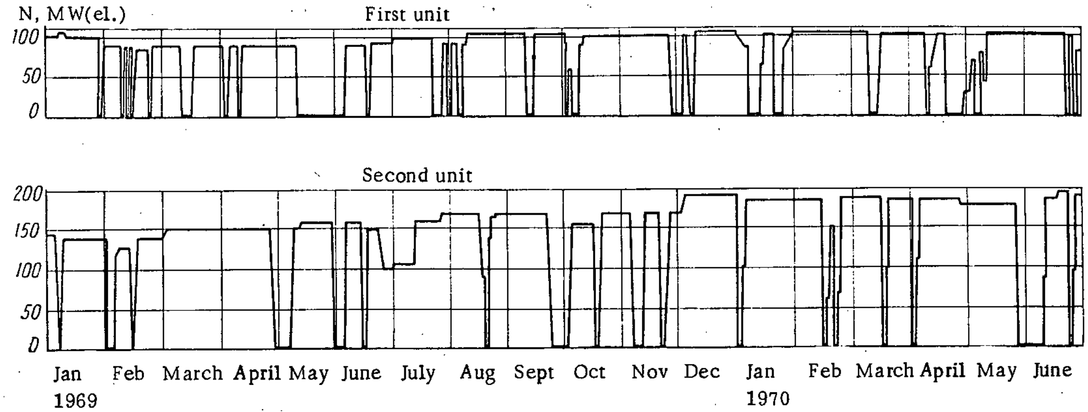

Fig. 3. Graph showing the loading of the first and second units of the Beloyarsk Nuclear Power Plant in 1969 and 1970.

The number of evaporation channels operating with superheated steam was increased in proportion to reactor power. The attained electrical power was ~75 MW with the evaporation circuit operating at full rated power.

The steam superheating channels were operated at full rated steam parameters in August 1967 and the electrical power was raised to 100 MW.

From start-up till July 1970 the first unit produced a total of 2.95 billion kWh of electric energy [7].

The second unit of the Beloyarsk Nuclear Power Plant was put into operation in late 1967 with an incomplete load of fuel channels. In November 1967, 453 evaporation channels and 138 steam superheating channels were loaded into the reactor, the superheated steam temperature reaching ~480°C. After the unit was started, fuel channels were added gradually in proportion to uranium burn-up until the full design number (732 evaporation and 266 steam superheating channels) was attained. This resulted in an increase of reactor power and steam temperature. With the reactor fully loaded a power output of 194 MW (el.) was attained in late 1967, the steam temperature being 515-520°C at the turbine input. Some of the steam superheating channels had an outlet temperature of 540-545°C [7].

Up to July 1, 1970 the second unit produced a total of 2.63 billion kWh of electric energy. The total power produced by the Beloyarsk Nuclear Power Plant up to July 1, 1970 was 5.58 billion kWh.

A large number of shutdowns during the initial period of operation of the first unit was caused by false operation of the safety system as a result of electrical circuit failures. In the course of operation we found it advisable to eliminate the emergency signals produced by fluctuation of coolant flow rate in the evaporation channels. This did not affect the reactor reliability while simplifying its operation. As more and more experience was gained in reactor operation other mass-flow control signals were removed from the safety system.

The permissible reactor power is essentially affected by the degree of uniformity of the radial distribution of energy-release fields in the core. Analysis of channel operation proved that the energy-release distribution along the core height is of secondary importance and that the thermal operating conditions of fuel elements change little as the nonuniformity ratio K_z varies between 1.4 and 1.5.

In the course of operation we have devised methods of measuring the energy-release fields with the aid of control rods, γ chambers, and charge sensors; the accuracy of measurements was also improved [8].

The actual radial and longitudinal energy-release fields in the core are close to the calculated ones. The radial field is in a great measure equalized by the control-system rods and by placing evaporation channels with richer uranium around the core periphery. Thus, the nonuniformity ratio of energy release from the evaporation channels was reduced to 1.21-1.23 in the second unit.

TABLE 1. Some Results of Operation of the Beloyarsk Nuclear Power Plant in 1969 and 1970

Description	First unit		Second unit	
	1969	Jan-June 1970	1969	Jan-June 1970
Number of scheduled maintenace shutdowns	17	6	8	5
Shutdown duration, h	1024	531	576	444
Number of other shutdowns	3	3	8	2
Duration, h	788	238	257	101
Time utilization factor, %	79,3	82,3	89,6	92,4
Power utilization factor, %	75,4	80,7	68,5	76,2

Good stability of radial energy-release fields and stable hydrodynamic characteristics of both the evaporation and the steam superheating sections made it possible to replace the individual flow rate control system by group control with the aid of throttling devices at the channel inlet.

Most representative of the performance of the Beloyarsk Nuclear Power Plant are the results obtained during 1969 and the first half of 1970. The graph in Fig. 3 shows loading of both units in the above-mentioned period; the number of shutdowns and their duration are listed in Table 1.

Scheduled maintenance shutdowns of the units were utilized for rearrangement and reloading of fuel channels in the reactor, extraction of faulty fuel channels, pressurizing the channels, calibration of flow rates, etc. The channel can be, as a rule, extracted from the reactor without any difficulties with an applied force of 300-500 kg; extraction of two-three channels takes 20-25 h including the time for shutdown and cooling of the unit and for returning it to full power.

The next step in the improvement and development of channel-type power reactors with tubular fuel elements are reactors with supercritical steam parameters. The experience gained in prolonged operation of the Beloyarsk Nuclear Power Plant confirmed the feasibility of steam superheating in the reactor and the possibility of operating the channels and fuel elements at high temperatures ($t = 540°C$, $t_{f.e.} = 630°C$). This formed a basis for the design of a reactor with supercritical steam parameters.

Transition to supercritical parameters with intermediate steam superheating allows a considerable gain in plant efficiency without affecting its main positive qualities: radiation safety, maneuverability, and the use of commercial power equipment. Moreover, supercritical steam parameters make it possible to increase the energy intensity of the reactor core, and to reduce the metal capacity of the reactor and of the power plant as a whole and thus reduce the cost of an installed kilowatt.

In the design of such a reactor, the steam parameters and thermal power are based on the operation of the reactor with two K-500-240 turbines. Thermal loading of the fuel elements and their geometry were designed to provide thermal operating conditions similar to those of the steam superheating channels of the second Beloyarsk unit.

The nuclear power unit with reactors designed to operate with supercritical steam parameters (see Fig. 1c) is a single-circuit design with preliminary steam superheating in an external intermediate superheater. Since the steam is superheated twice in the reactor all fuel channels are divided in accordance with their purpose into three groups and are connected in series along the coolant path. Feed water flowing into the reactor after passing through high-pressure regenerative preheaters is heated in the preheater channels to 400°C and through the outlet and distribution headers passes into the primary superheating channels where it is heated to 560°C. Superheated steam from the primary superheating channels is directed to the intermediate superheater where it transfers its heat to the steam used-up in the high-pressure cylinder of the turbine. After passing through the intermediate superheater, the coolant flows into the mixer which serves also as a superheating regulator and then through the distributing header to the secondary superheating channels. From the secondary superheating channels, steam heated to 540°C is directed to the turbine.

The heat flow diagram with intermediate mixing allows a more uniform distribution of the increase of heat content in the primary- and secondary-superheating channels and enables independent regulation of superheated steam temperature at the steam superheating channel outlet by varying the flow rate of feed water into the preheating channels and mixer.

Supercritical coolant parameters make it possible to eliminate the multiple circulation loops with steam separators and circulation pumps, but require a pressure of at least 240 abs. atm to be maintained in the reactor in all start-up, transient, and operating conditions of the unit. Reduction of pressure below critical can cause flow pulsation in individual channel branches and result in fuel element burn-out, in particular with low volumetric flow rates and relatively high heat flows in the channels. This applies to all three groups of fuel channels.

In developing the start-up scheme we have taken into account the experience gained in power plants with supercritical steam parameters operating with organic fuel. After considering all sorts of start-up schemes, we have chosen a scheme with an external start-up separator and an ROU* system by means of which the pressure in the circuit is maintained within allowable limits (not below 240 abs. atm).

During coolant and equipment heat-up the coolant flow rate through all fuel channels is set to 25-30% of nominal. Calculations indicate that such a flow rate ensures reliable nonpulsating channel cooling in the entire range of reactor heat-up temperature up to the moment the operating parameters are reached.

After the reactor the coolant is throttled and flows into the start-up separator. From the separator, the steam at a pressure of about 20 abs. atm is directed to the intermediate superheater and then through the ROU into the turbine condenser. Some of the steam can flow into the low-pressure preheaters, to the feed-water deaerators, and high-pressure preheaters, and this speeds up the equipment heat-up and reduces heat loss during the start-up period.

From the start-up separator water flows into the deaerator vessels which improves deaeration of feed water during start-up.

When the heat content of the coolant at the reactor outlet exceeds the heat content of steam in the saturation lines at 20 abs. atm, the steam is directed to a BROU† whence it enters the intermediate superheater and the regenerative preheaters.

Preheating of steam lines leading to the turbine takes place during this period. The turbine starts when the temperature at the reactor outlet reaches 540°C.

Steam temperature at the secondary-superheating channels is maintained within allowable limits by appropriate flow rate of feed water to the mixer. The temperature at the primary-superheating channels outlet increases with increasing reactor power. At the same time, all the steam from BROU is gradually transferred to the turbine.

Preliminary superheating takes place in the intermediate superheater by means of steam coming from the primary-superheating channels so that the intermediate-superheater temperature rises with increasing reactor power. Thus, the turbogenerator and the entire plant equipment is in operation when the reactor power reaches $\sim$30% of nominal.

Underheating of steam in the intermediate superheater during start-up can be tolerated as the load on the turbine is slight during this period and an increased amount of moisture content in steam in the last turbine stages is allowable.

An electric pump with a capacity of about 30%, used for start-up and as a stand-by, operates during equipment heat-up till superheated steam is obtained at the reactor outlet. The turbopump, fed by steam from the ROU at $\sim$16 abs. atm, takes over as soon as superheated steam is available at the reactor outlet.

This start-up scheme allows the start-up equipment to be used also for reactor cooling and shutdown. The sequence is then reversed, the operation is transferred from the turbine to the BROU, and then as the power is reduced, to the ROU and the start-up separator. The circuit heat is carried off to the turbine condensers; the steam flows into these condensers from the ROU or, in the case of deeper cooling, from the deaerators.

As during heat-up, pressure in the reactor circuit in the course of cooling is maintained at 240 abs. atm and can be reduced only when the temperature at the reactor outlet is lower than the saturation temperature at the given pressures. During cooling, the flow rate of feed water is maintained at 30% of nominal capacity of feed pumps.

In case of load dump, BROU is automatically connected and steam flows to the feed-pumps drive turbine not from the intermediate-pressure cylinder but from the steam discharge line after the BROU. The reactor feed is thus practically uninterrupted in the case of a load dump.

From the above follows that the start-up and apparently also continuous operation of nuclear power stations with supercritical steam parameters will differ only slightly from the start-up and operation of the Beloyarsk Plant and conventional thermal plants with supercritical parameters.

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* ROU – Pressure reducing and cooling unit.
† BROU – Rapid reduction and cooling unit.

On the basis of the experience gained in the design, construction, and operation of the Beloyarsk Nuclear Power Plant with a channel-type, uranium – graphite power reactor with tubular fuel elements and nuclear steam superheating one can draw the following conclusions:

1. The problem of nuclear steam superheating and the use in nuclear power engineering of regular industrial turbines as well as conventional thermomechanical and electrical equipment has been positively solved.

2. A technique was devised for independent reactor start-up in about 12 h with minimum special-purpose equipment.

3. Gross efficiencies obtained at the Beloyarsk Plant are 36.9% in the first unit and 37.8% in the second unit.

4. The cost of electric energy planned for 1970 is 1.188 kopecks per kWh (the design cost was 1.230 kopecks per kWh).

5. Single-circuit channel-type uranium – graphite reactors are advanced constructions from an engineering point of view.

6. Because of the use of tubular fuel elements with single-sided heat removal such reactors are highly radiation safe both with respect to the operating staff and to the population living in the area.

7. The feasibility of using special-purpose fuel channels (for example for steam generation and superheating) open up possibilities for the design of highly efficient, high-power nuclear plants with subcritical, and in particular with supercritical parameters, nuclear steam superheating, and a unit capacity up to 2 million kW, that are convenient in operation and economical in capital expenditure.

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RADIATION SAFETY IN THE DESIGN AND OPERATION OF CHANNEL POWER REACTORS

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The successful development of nuclear power is due not only to the fact that atomic power plants, even in their present stage of development, compete economically with ordinary thermal installations, but also that operating experience has shown that atomic power plants do not produce any essential alteration of the radiation environment in their vicinity [1, 2]. Operating experience with Soviet atomic power plants using graphite-moderated channel type reactors cooled with ordinary water [3, 4] permits the analysis of certain aspects of radiation safety assumed during their development [5]. The analysis presented below is based on operational and research data obtained at the I. V. Kurchatov Beloyarsk Atomic Power Station (BAPS).

Shielding against Reactor Radiation

The standard practice in the USSR of placing the reactor and main equipment in a concrete building in a strictly controlled zone is economical and ensures the normal and safe operation of the atomic power station. At an atomic power station with channel reactors the side shield (the concrete reactor pit) is generally one of the main structural elements of the building. An annular water tank serves as part side shield and part support structure. The water tank lowers the radiation flux on the concrete and decreases the temperature of the reactor vessel. This permits the use of ordinary structural concrete for shielding and low-melting-point steel for the vessel. Present atomic power station design, based on experiment, tends toward the wide use of load bearing box-like metal structures with spaces for holding shielding materials such as rock, minerals, ore, and sand [6-8] which are relatively inexpensive, convenient to place, and have good shielding properties. This reduces the shielding volume and the over-all dimensions of the reactor building.

In order to provide access to the channels from above, the shield over the top of the reactor is ordinarily made in sections. It is not only a shield against core radiation and capture γ-rays from shielding materials and coolant, but also serves as the floor of the reactor room. In designing the top shield particular attention must be paid to inhomogeneities such as gaps, channels, slits, connections, etc., and to radiation streaming. All of these make significant contributions to the total dose rate above the reactor.

Alongside the reactor pit are rooms (boxes) containing technological equipment and coolant loop facilities. Until recently the shielding of these rooms against reactor radiation has been designed to ensure a certain dose. Operating experience at atomic power stations of the channel type has shown that the total

TABLE 1. Activity of Coolant in Second Unit, Ci/kg

Sampling location	Co^{60}	Zn^{65}	Mn^{54}	Cr^{51}
Separator scavenging water	$(1-3)\cdot 10^{-7}$	$(4-8)\cdot 10^{-7}$	$(0,4-3)\cdot 10^{-8}$	$(0,3-3)\cdot 10^{-7}$
Saturated steam of separator	$(0,5-1,5)\cdot 10^{-8}$	$(1-3)\cdot 10^{-8}$	$(0,3-5)\cdot 10^{-9}$	$(0,9-2)\cdot 10^{-8}$
Live steam	$(0,2-2)\cdot 10^{-8}$	$(0,1-1)\cdot 10^{-8}$	$(0,1-5)\cdot 10^{-9}$	$(0,2-5)\cdot 10^{-8}$
Steam condensate beyond turbine	$(0,3-5)\cdot 10^{-9}$	$(0,4-3)\cdot 10^{-9}$	$(0,2-1,5)\cdot 10^{-9}$	$(0,5-6)\cdot 10^{-9}$

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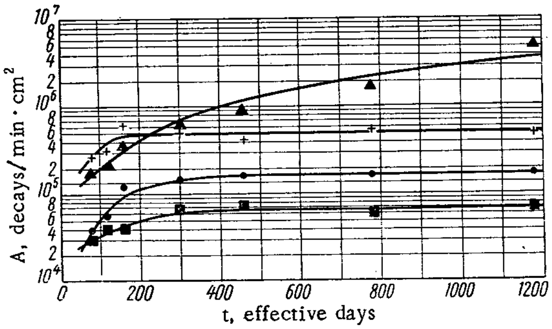

Fig. 1

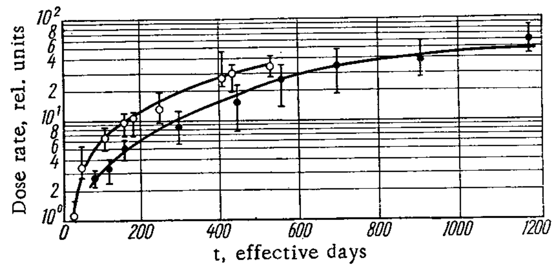

Fig. 2

Fig. 1. Activity of deposits A on inlet piping of evaporative loop of first BAPS unit as a function of operating time t: ▲) Co^{60}; ■) Co^{58}; ●) Mn^{54}; +) Cr^{51}.

Fig. 2. Relative change in after-shutdown dose rate close to equipment of evaporative loop as a function of operating time: ●) first BAPS units; ○) second BAPS unit.

after-shutdown dose rate in rooms with power equipment is determined by the radiation from radioactive corrosion products deposited on the inner surfaces of the coolant circuit. Noting that shielding costs are at least 30%, and labor not less than 40% of the total construction cost of an atomic power station [9], it is expedient to base the shield thickness of the rooms containing the coolant loop equipment on the dose rate produced by the radiation from radioactive corrosion products.

In the last few years there has been a tendency to do away with the side water tanks in order to lower operating expenses. Instead of water the tanks can be filled with rocks and minerals which retain their water of crystallization at high temperatures [10, 11], or with radiation- and heat-resistant concrete [12, 13].

Coolant Activity, Radiation Environment, and Emissions at an Atomic Power Plant

One of the most important aspects of radiation safety in an atomic power plant is the contamination of the environment by radioactive emissions both during normal operation and in case of an accident. The main sources of radioactivity of the coolant are its own activity, the activity of corrosion products and admixtures, and the activity due to the entrance of fission products into the coolant. We discuss these component activities on the basis of operating experience with the channel reactors of the Beloyarsk Atomic Power Station (BAPS).

It is well known [5] that the BAPS consists of two units developing 100 and 200 MW of electric power. It is the first experimental-industrial nuclear power plant using nuclear superheated steam. The main difference between the two BAPS units is in their thermal systems; the first unit has a two-loop system and the second a single loop. In view of the relatively small amount of coolant and its short residence time in the core, the amount of self activity of the coolant in these reactors is intrinsically smaller than in boiling-water reactors of the pressure vessel type, and consequently the radiation environment close to equipment such as the turbine is more favorable. For this same reason the rate of formation of radiolytic gases is lower, and this decreases the corrosiveness of the steam going to the turbine and regenerative heaters and also decreases the amount of uncondensed gases exhausted by the turbine ejector pumps. The activity of the exhaust gases is determined by N^{13}. The amounts of N^{13} produced per unit of thermal power developed in the first and second BAPS units are respectively 200 and 10 times smaller than in a pressure vessel type boiling-water reactor at Garigliano [14]. It is noted also that the rate of production of radiolytic gases in the BAPS reactors is only a fifth as large as in a pressure vessel boiling-water reactor of the same power. The yield of radiolytic oxygen in the second BAPS unit is 0.25 ± 0.05 molecules per 100 eV of radiation absorbed by the coolant.

An important factor in the operation of an atomic power plant with a single-loop system for steam generation is the radiation level near the turbine. During operation the intensity of γ-radiation is relatively low near both BAPS turbines operating with steam superheated in the reactors. The dose rates measured near the high pressure cylinders at nominal power are respectively 1-2.5 and 4-10 μR/sec for the first and second units, and near the low pressure cylinders 0.3-2 and 3-8 μR/sec. As is well known the BAPS uses

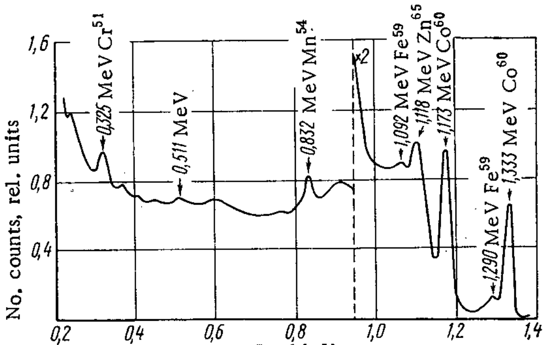

Fig. 3

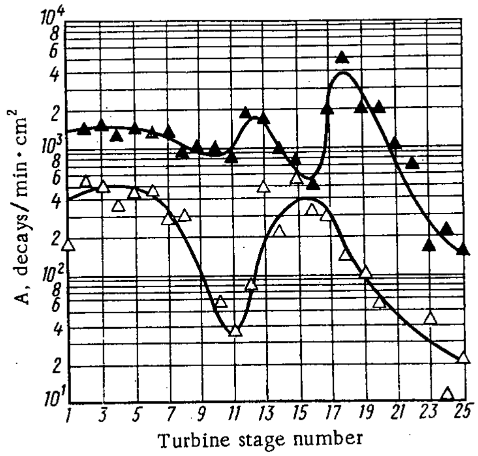

Fig. 4

Fig. 3. Gamma-spectrum from radioactive deposits on regenerative heater of the second BAPS unit measured with a Ge(Li) detector.

Fig. 4. Specific activity of Co^{60} deposits on turbine blades after 160 (Δ) and 460 (▲) effective days of operation of first BAPS unit.

standard type VK-100-6 turbines without intermediate reheaters and steam separators [15]. No shielding is provided against radiation from the turbines.

Operational experience with power reactors in various countries has shown that the radioactivity of corrosion products deposited on the surfaces of eqiupment and piping does not represent a danger in the full sense of the word but it is the main source of radiation to station personnel performing routine preventive maintenance and determines the total radiation dose received by power plant personnel [16]. The problems of corrosion activity at the BAPS have been dealt with repeatedly in the press [17-19] and therefore we present only the most interesting results.

The coolant in the first BAPS unit is relatively free of long-lived radioactive corrosion products. The primary loop contains up to 10^{-8} Ci/kg and the secondary loop up to 10^{-9} Ci/kg. Table 1 shows the activity due to long-lived isotopes in the coolant of the second unit at various sampling locations. The coolant activity in the first and second units differ both in composition and in magnitude because of the difference in coolant loops and because of the higher oxygen content of the steam – water mixture and the steam in the second unit. It should be emphasized that the low activity of live steam shows the possibility of using a power plant of this type for central heating without an intermediate loop.

Thus the activity, even in live steam of the second BAPS unit, of isotopes whose concentrations in water are regulated by the "Health rules" [20] is no more than twenty times the maximum permissible concentration in water of open reservoirs and water works.

Figure 1 shows data on the activity of deposits at one of the typical locations in the evaporative loop (at the reactor inlet) of the first BAPS unit, and Fig. 2 shows the relative change in dose rate close to equipment of the evaporative loops of the first and second units as a function of operating time.

A typical γ-spectrum from radioactive deposits on the regenerative heater of the second BAPS unit is shown in Fig. 3.

Figure 4 shows the distribution of Co^{60} activity on the turbine blades of the first unit as a function of operating time, and Fig. 5 shows the distribution of Cr^{51} and Zn^{65} activities on a turbine of the second unit.

The dose rate close to primary loop equipment of the first BAPS unit, measured at shutdown after 1180 effective days of operation, was 2-200 μR/sec; near the secondary loop equipment it was 0.1-2.2 μR /sec. The dose rate near equipment of the circulating part of the loop of the second BAPS unit at shutdown after 530 effective days of operation was 2-100 μR/sec, and near equipment of the condenser feed circuit it was 0.6-20 μR/sec. In spite of the fact that the BAPS fuel elements are clad with stainless steel, the radiation level close to the primary loop equipment of the first unit is somewhat lower than at a power station with pressurized water reactors (for identical effective operating times).

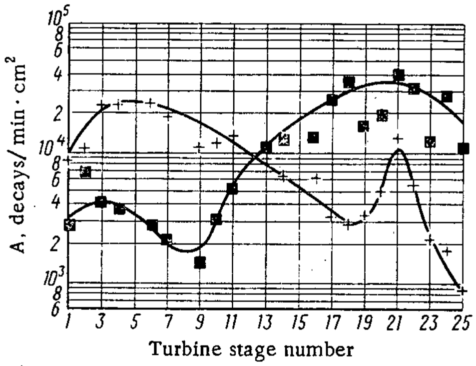

Fig. 5

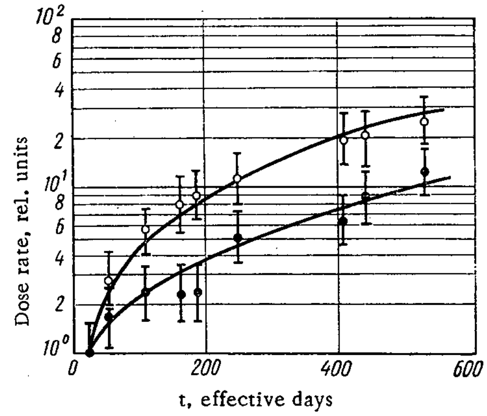

Fig. 6

Fig. 5. Distribution of Cr^{51} (■) and Zn^{65} (+) activities on blades of turbine No. 2 of second BAPS unit after 294 effective days of operation.

Fig. 6. Relative change in dose rates close to equipment of steam condensing and feed circuits of the second BAPS units filled (○) and not filled (●) with water from the evaporative part of the loop in startup and cooling procedures.

Operating experience with the first BAPS unit has shown that the radiation dose rates near the secondary loop equipment, including the turbine, while maintenance work is going on practically do not limit the servicing time, i.e., the technological scheme of the first unit is very successful from the point of view of ensuring radiation safety and for providing for the servicing of the steam condensing circuit.

It has been established that filling parts of the steam condensing and feed circuits of the second unit with water from the evaporative part of the loop for cooling and startup operations leads to a significant increase in the activity of the corrosion deposits on inner surfaces (Fig. 6).

One of the main features of the I. V. Kurchatov BAPS reactors which has been noted frequently [21] is the tubular construction of the fuel elements with cooling from one side, eliminating contamination of the coolant by fission products for practically any damage to the fuel channels. This eliminates the escape of fission products into the loop, to the turbine, and into the atomosphere with the exhaust gases.

Still another important advantage of channel reactors should be noted – the possibility of detecting a ruptured fuel element in time and rapidly replacing a dangerous channel. In principle this can be accomplished even on an operating reactor. These operations are very difficult in reactors with pressure shells and generally require a lengthy shutdown of the power plant.

To localize the effects of accidents in atomic power plants with reactors of the pressure vessel type, particularly accidents involving the rupture of a loop and the escape of fission products, the reactor and coolant loop equipment are sometimes enclosed in a containment vessel. Arranging a containment vessel around a large power reactor and all the equipment and facilities for a single-loop atomic power plant, is a rather complicated problem whose solution requires using more materials. In this sense channel reactors, particularly those with tubular fuel elements, do not have to be placed with equipment in a special containment vessel since fission products are contained not only by the fuel element tubes but also by the reactor vessel around which there can be a second shell forming a water tank, and by metal structures.

Two types of channel accidents are possible with the BAPS tubular fuel elements – "dry" and "wet". In a "dry" accident only the outer cover of the fuel element is ruptured. In this case mainly radioactive noble gases (RNG) enter the gaps in the graphite reactor stack which are filled with pure nitrogen at a gauge pressure of 5-20 mm of water. In a "wet" accident the inner tube of the fuel element is ruptured and the coolant enters the graphite stack where some of it is vaporized. The steam – gas mixture is withdrawn into a system of accident localization gas containers where the steam is condensed and the RNG are held for 24 hours and purged of radioactive aerosols by thin-fibered and carbon filters. Extensive operational data on the two BAPS units show that under normal operating conditions the direct ejection of RNG into the atmosphere is 20-80 Ci/day for the first unit and 15-60 Ci/day for the second. Rupture of a fuel element increased the escape to 150-200 Ci/day, with no value as large as 400 Ci/day. Thus the total activity of RNG escaping from both units is below the health norms by a factor of 25-100. The escape of radioactive aerosols is due mainly to corrosion products and amounts to 10^{-3}-10^{-4} Ci/day. The exhausts from the turbine ejector pumps do not contain RNG. Their activity is determined mainly by N^{13} and therefore they are vented into the atmosphere without holding or special scrubbing.

A consideration of maximum possible accidents such as the rupture of the main pipelines on channel type reactors with tubular fuel elements shows that the scram system and the emergency cooling system can prevent massive melting of fuel elements and the growth of the accident, due largely to the thermal capacity of the graphite stack. In addition, because of the high thermal conductivity of graphite and the channel construction, the heat from the damaged channel is transferred to the graphite and carried off by adjacent channels. The existing control system enables the damaged channel to be detected and replaced in time. All this helps to prevent the escape of fission products in an accident from becoming a serious danger.

Operating experience with channel power reactors in the USSR has confirmed the validity of this approach to the solution of radiation safety problems and of the principles underlying the design of reactors of this type.

One of the main advantages of these reactors over pressure vessel reactors is the possibility of detecting a ruptured fuel element in time and replacing it quickly without shutting down the power plant. The use of tubular fuel elements in a channel reactor prevents the entrance of fission fragments into the coolant loop and localizes the fission fragment activity from a ruptured fuel element in a gas at low pressure.

In this sense the operating experience at the I. V. Kurchatov BAPS offers convincing proof that the population and the servicing personnel are not endangered by radiation from reactors of this type, and indicates the possibility of constructing such power stations near large cities and using them for central heating.

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CONSTRUCTION OF URANIUM – GRAPHITE CHANNEL-TYPE REACTORS WITH TUBULAR FUEL ELEMENTS AND NUCLEAR-SUPERHEATED STEAM

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UDC 621.039.52

On June 27, 1954, at the Obninsk atomic power station, electric current was generated from the energy of fission of the atomic nucleus for the first time in the world in a uranium–graphite reactor of the channel type with tubular fuel elements.

In designing the first-ever atomic power station, we were faced with a problem: the construction of a reliable industrial power station and the creation of a basis for training personnel.

The reactor of the "First Atomic Power Station in the World," using thermal neutrons with a graphite moderator and high-pressure water as a heat-transfer medium, has a thermal power of 30 MW. It uses a two-circuit heat system: water in the first circuit, circulating through the fuel channels under a pressure of 100 atm, transfers heat in a steam generator to the water in the second circuit, which is vaporized and drives a turbogenerator with a power of 5 MW.

After one year's operation of the "First Atomic Power Station in the World," conditions were laid down for the further development of nuclear power production in the USSR.

The drive to make cheaper atomic power station equipment led to the creation of reactors with nuclear steam superheating and a new type of design solution for the thermal circuit, originally two-circuit with superheating of steam in the second circuit [first block of Beloyarsk I. V. Kurchatov atomic power station (BAES), 100 MW (el.)], and later single-circuit [second block of BAES, 200 MW (el.)]. In both BAES blocks there were production-line turbines working at 90 atm and 500-535°C. A further development of these systems is a single-flow circuit with a supercritical reactor (power 800-1000 MW (el.)) with production-line turbines working at 240 atm and 540°C [1-3].

TABLE 1. Some Characteristics of the Fuel Channels

Characteristic	First block of BAES		Second block of BAES		Planned		
	EC	SHC	EC	SHC	CPH	C1H	C2H
Dimensions of interior tube of fuel element, mm	9,4×0,6	12×0,6	12×0,6	12×0,6	11×0,8	11×0,8	12×0,8
Dimensions of fuel can, mm	20×0,2	20×0,3	20×0,2	20×0,3	18×0,3	19×0,3	16×0,3
No. of fuel elements in channel	6	6	6	6	8	8	8
Dimensions of central tube of channel, mm	18×1	—	20×1	—	—	—	—
Maximum power of channel, kW	408	326	623	729	2660	1800	2130
Main parameters of channels (P_{out}/T_{out}, atm/deg C)	$\frac{135}{330}$	$\frac{105}{520}$	$\frac{130}{330}$	$\frac{100}{520}$	$\frac{300}{400}$	$\frac{290}{545}$	$\frac{280}{565}$
Weight of channel, kg	200	200	200	200	300	300	300

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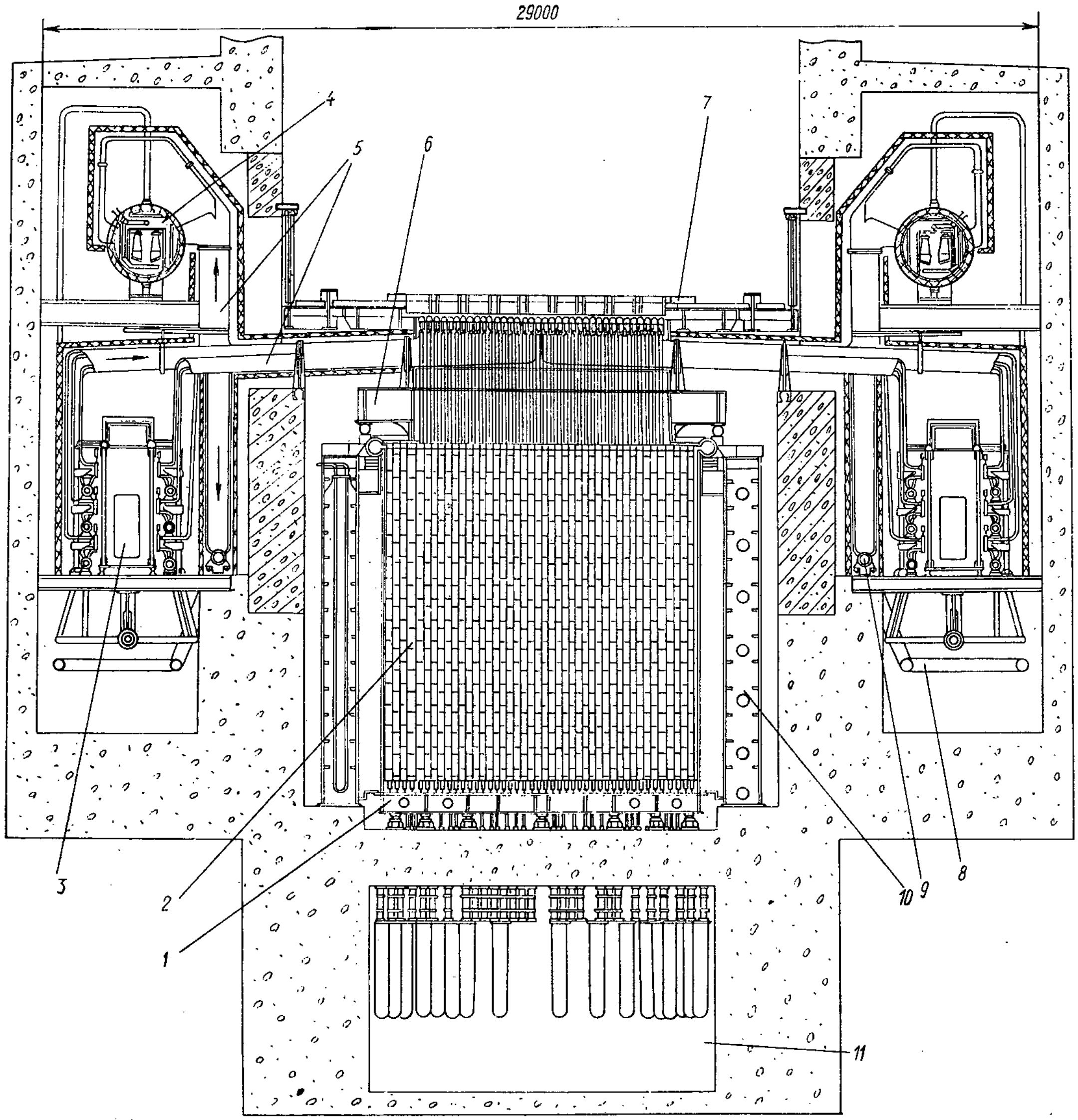

Fig. 1. Longitudinal cross section of reactor of first or second block of BAES. 1) Bottom plate; 2) graphite stacking; 3) corridor for servicing equipment; 4) steam separator; 5) ducts; 6) top plate; 7) upper shuttering; 8) main ducts; 9) collector for superheated steam; 10) water shielding; 11) room for control rod drives.

Constructional Features of BAES Reactors

At present there are two reactors of the channel type in industrial service at BAES (one since 1964, the other since 1967). They consist of the following main units (Fig. 1): evaporator and superheater channels (EC and SHC), graphite stacking (moderator and reflector), top and bottom plates, can, biological shielding tank, upper shuttering, separator drums, and ducts for incoming and outgoing coolant to the channels.

To superheat the steam to 500-520°C it was not only necessary to build channels in which boiling and superheating could take place, but also to make another approach to the construction of the graphite stacking and the metal structures of the reactor.

It was necessary to solve problems concerning compensation of temperature expansion between the top and bottom plates, between the stacking vessel and the plates, between the channel riser pipes and the graphite stacking columns, between the graphite blocks, and so forth. In choosing the sizes of the gaps between stacking columns account was taken of thermal expansion of the graphite blocks and swelling of

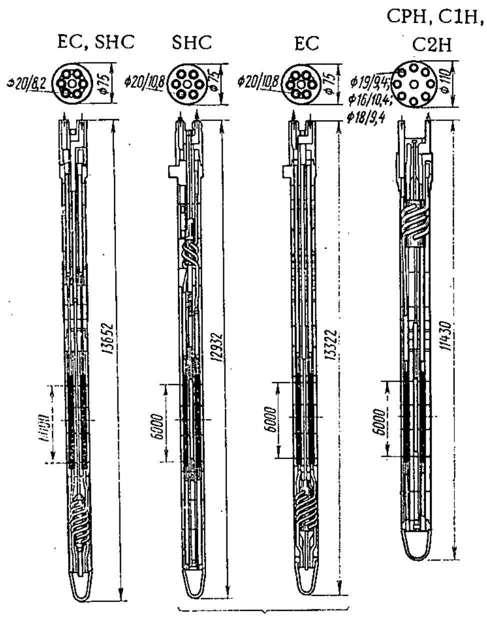

Fig. 2. Construction of fuel channels. a) Reactor of first BAES block; b) reactor of second BAES block; c) reactor with supercritical steam parameters (project).

the graphite. Analysis of the above factors led to decisions on clearances in the manufacture of the graphite blocks, in marking out the spacing between holes in the top and bottom plates, etc.

The temperature of the graphite stacking depends markedly on the gaps between a fuel element and the hole in the graphite sleeve of the channel, and between the hole in the block and the channel sleeve. The gap size itself depends directly on the precision of manufacture of the fuel elements, sleeves, and blocks. The graphite stacking blocks were made with tolerances of Class 3-4, which meant that the temperature of the exterior surface of a block did not exceed 750-800°C.

The Metal Parts of the Reactors. These consist of separate sections, convenient for production, transportation, and installation.

Constructional decisions were checked experimentally during the preliminary and detailed planning stages. The following research was most important.

1) Experimental determination of the voltages in individual units of the metal structures and the components of the fuel channels, based on optical polarization in brittle lacquer and tensometry.

2) Checking that the channels will admit the graphite columns in working conditions.

3) Checking the channels for stability.

4) Checking the efficiency of the channel compensators in working conditions, etc.

The designs of the reactors in the first and second blocks of the BAES were similar and of equal size, and differed essentially only in the arrangements of the EC and SHC.

The graphite stacking was assembled on the bottom plate and consists of individual graphite blocks 200 × 200 mm in cross section, which when assembled form vertical columns with central holes for the fuel channels (998) and channels with automatic control rods (6).

The channels for the manual control rods (78) and scram system (16) were located between the fuel channel blocks. The drives to the control rods were located below the reactor.

The stacking is enclosed in a sealed vessel, welded directly to the bottom plate and, via a thermal expansion compensator, to the top plate. The top and bottom plates are metal box-built structures with an upper bearing plate 42 mm thick fixed by vertical ribs. The height of the top plate is 1.5 m, that of the bottom plate 0.6 m. To the vertical ribs of the bottom plate is welded a plate forming the seal for the water-filled cavity. To remove heat, within the plate below its top sheet is wound a coil through which cooling water circulates.

Behind the stacking vessel is the lateral biological shielding, consisting of a water tank 1 m wide and a concrete shield 3 m thick. The top plate rests on the water tank, and between its vertical pipes are ducts for the incoming and outgoing coolant.

Circulating water to the EC and saturated steam to the SHC are collected from distributor heads on either side of the reactor. Here we also find the collectors for the superheated steam.

On the pipes from the collectors are control valves and rotameter-type flowmeters. The pipes carrying steam – water mixture to the separator and superheated steam to the collector are fitted with valves, the pipes from the EC with surface resistance thermometers, and the pipes from the SHC with thermocouples.

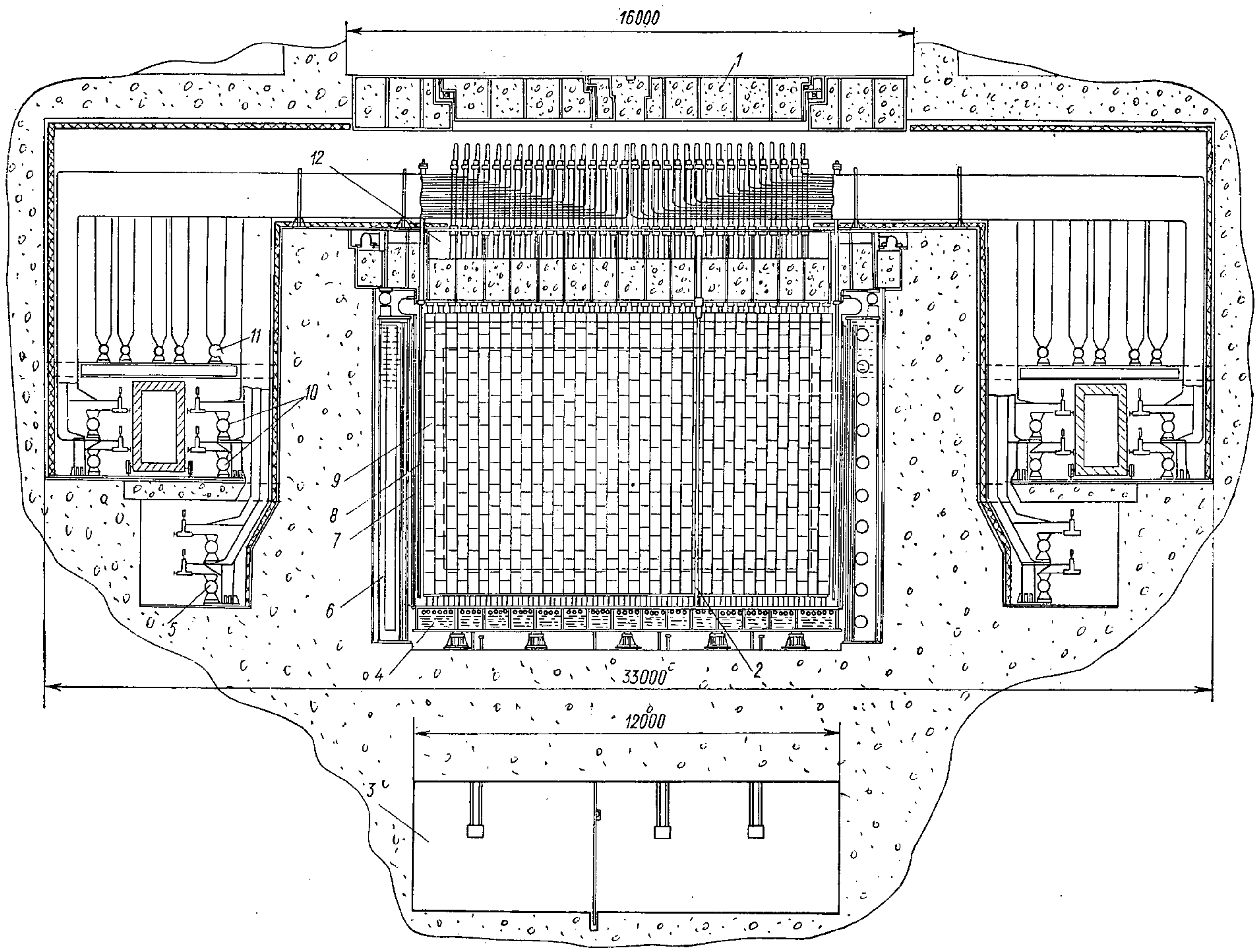

Fig. 3. Longitudinal cross section of reactor with supercritical steam (projected): 1) top shuttering; 2) fuel channel; 3) room for safety control rod drives; 4) bottom plate; 5) pressure collector; 6) biological shielding tank; 7) vessel; 8) thermal screen of vessel; 9) graphite stacking; 10) pressure collectors of first and second superheats; 11) collector for superheated steam; 12) top plate.

Damage to the external tubes of the fuel elements is monitored by activity of the gas. Gas is pumped along the channel gap round the fuel elements by a specially-designed can sealing monitor system (CSMS). With the aid of the same system, breakages of ducts are detected by means of the rise in pressure.

Construction of Channels. The EC of the first and second blocks were of similar design (Fig. 2); in external appearance they are cylinders, 75 mm in diameter, consisting of tubes, graphite sleeves, and fuel elements. Coolant from the duct pipe passes through a junction manifold and a branch pipe to the upper channel head and down through the central tube to the shaft. From the shaft, through throttle disks and tubular compensators which take up the different thermal expansions of the down and up tubes, coolant flows up over the internal surface of the fuel element tubes. In the EC the water is heated to boiling and partly evaporated.

From the fuel elements the coolant flows through a large-diameter tube to the top channel head and then through a side tube and junction manifold through a pipe to the separator drum.

Table 1 lists the dimensions of the fuel elements.

The SHC used at present in the BAES is a design of six fuel elements and steel tubes in metal and graphite sleeves forming a cylinder of 75 mm diameter.

Coolant passes through the inlet pipe at the head of the channel into three fall branches and through a tail collector chamber to three rise branches and then into a collector chest, from which it passes through

TABLE 2. Main Characteristics of Reactors

Characteristic	First block of BAES (730 EC, 268 SHC)	Second block of BAES (732 EC, 266 SHC)	Project (264 CPH, 358 C1H, 400 C2H)
Reactor power, MW (el.)	100	200	800-1000
Total metal content (top and bottom plates, vessel, biological shielding tank, etc.), tons	1800	1800	~ 2000
Weight of separator drums, tons	94	156	-
Weight of circulation loop, tons	110	110	370
Weight of graphite stacking, tons	810	810	1200

annular gaps between the 35 × 2 mm and 17 × 2.5 mm tubes through a junction chamber and a 22 × 1.8 mm tube into the top outlet branch. In the down branches of the SHC are tubular compensators which take up the different thermal elongations of the up and down branches.

The successful operation of the BAES shows that the reactors in this station have potentialities for increased power.

Superheating of the steam in the reactor permits the use of production-line steam turbines of good characteristics, made in the USSR, for an atomic power station.

Construction of Reactor with Supercritical Steam Parameters (Plan)

A further development of uranium–graphite reactors of the channel type is a reactor with supercritical coolant parameters, permitting the use of a single-flow circuit in which there is no separator drum or circulation pump.

The distinguishing feature of the reactor with supercritical steam is the high electric power (up to 800-1000 MW) from an active zone (core) of practically the same size as those of the BAES reactors.

The metal structures (Fig. 3) duplicate the design solution of the existing BAES reactors.

The reactor stacking, 7.5 m in height and 11.5 m in diameter, is made up of separate graphite blocks of hexagonal shape with holes 110 mm in diameter. The thickness of the top, bottom, and side reflectors is about 0.6 m. Above and below the columns of graphite blocks are cast-iron blocks. The height of the core is 6 m and its diameter is 10.2 m.

The reactor has three groups of fuel channels – heating (CPH), first superheat (C1H), and second superheat (C2H). The heating channels are in the center of the core, while the second-superheat channels encircle them and the first-superheat channels are on the periphery. The channels are of identical construction and differ only in the materials and dimensions of the fuel elements.

The fuel channel, shown in Fig. 2, consists of an input distribution chamber to which are welded four down tubes with fuel elements. In the bottom part of the channel is a tailpiece with the down and up tubes welded in.

The four up tubes with their fuel elements in the upper part of the channel are welded to the collector chamber. The fuel elements are located in holes in the graphite sleeves.

In the down tubes are coiled thermal expansion compensators. The channel is connected by welding to the rise pipes of the top plate and to the inlet and outlet pipelines.

The sealing of the fuel cans and the integrality of the channel tubes are monitored by pumping nitrogen through special ducts in the channel and by pulse tubes in the control system with appropriate instruments.

The use of sectioned graphite sleeves in the fuel channels permits reduction in the gap between the channel and the stacking blocks, improving heat output from the graphite and reducing the temperature of the stacking.

The drives to the control rods are located below the bottom plate and biological shield. This facilitates servicing of the heads of the fuel channels during readjustment or replacement of spent channels.

The use of a new design for the control rods and the arrangement of parts of the biological shielding in the top and bottom plates reduces the length of the fuel channels and somewhat reduces the height of the building.

Since the spacing of the fuel channels was increased, the size of the graphite stacking blocks was also increased. The measuring instruments in the channels are inspected from a mobile cabin.

Above the heads of the fuel channels is a hinged or removable shuttering, which simplifies the process of recharging the channels and permits inspection of the channel heads and of the inlet and outlet pipes during routine maintenance. The shuttering is moved by a gantry crane in the central room.

The grouping of the main equipment of the reactor with supercritical steam, the separation of the ingoing and outgoing piping, the position of the pressure and collector headers, and the arrangement of the control and flowmeter systems differ advantageously from those in a reactor with ordinary high coolant parameters, primarily owing to the elimination of the steam separator drums and the circulation pumps.

Table 2 gives the main characteristics of the above reactors.

* * *

From an analysis of the above designs for uranium – graphite reactors of the channel type with tubular fuel elements and either subcritical or supercritical steam, it follows that the scale of the metal structures (top and bottom plates, stacking vessel, side shielding water tank, top shuttering, and graphite stacking) does not depend on the coolant parameters; the thermal emission and the operating temperatures of the metal structures do depend slightly on the reactor power.

Unlike reactors of the vessel type or of the channel type with tubes under pressure and rod-type fuel elements, the use of tubular fuel elements permits construction of a single standard reactor design in which the channels and steam – water pipes are changed.

Practically no special steels are required for the metal structures of the reactors; ordinary boiler-making materials are quite adequate. The construction and assembly techniques are simpler than those of steam boilers.

The reactors can be built in a large number of medium-size factories.

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MANAGEMENT OF WATER AND CHEMICALS AT NUCLEAR POWER STATION WITH CHANNEL TYPE REACTOR AND NUCLEAR STEAM SUPERHEAT

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The uranium–graphite boiling-water reactor of the second power generating unit in the I. V. Kurchatov Belyi Yar nuclear power station (BAES) operates on a single-loop thermal flowsheet. The steam generated in the reactor is sent directly to the turbines, and the steam condensate provides the main component for the reactor feedwater.

The structural material of the basic reactor components and subsystems with which the coolant (water and steam) comes into contact is 0Kh18N10T stainless steel. Channels and fuel-element cladding, as well as the piping for the reactor circulation loop, are made of this stainless steel. The separator drum and the piping for the live steam are made of pearlitic steel.

The reactor operates with two standard VK-100 turbines. All the turbine components, as well as the regenerative preheaters, are made from the standard materials commonly in use in the design of turbomachinery. The vessel and the PVD [high-pressure steam] piping, the lines taking bled steam to the preheaters, the vessels of the deaerators, condensers, PND [low-pressure steam] vessel, condensate lines, and feedwater lines, are all made of pearlitic steels. The PND tube furnace is made of brass, and the condenser tubing is made of MNZh-5-1 alloy.

About 15% of the surfaces of the loop are made of 0Kh18N10T type steel, and about 25% of pearlitic steels, while the remaining surfaces are made of brass and MNZh-5-1 alloy.

When different structural materials are used, with the presence of rather appreciable surface areas of pearlitic steels, and with steam generated in a boiling-water reactor, water management conditions must be arranged so as to minimize the rate of corrosive attack on the structural materials, and to minimize deposits and crud on the surfaces of the fuel elements.

The basic difficulty in the way of attaining such optimum conditions stems from radiolysis of the water. Water radiolytic processes occur most intensively in boiling-water reactors, as has been learned. The underlying reason is that molecular radiolysis products (oxygen, hydrogen) are removed with the steam from the water before the reactions involving recombination of the molecular radiolysis products and the resulting radicals have time to go to completion. The limiting value of the hydrogen yield in radiolysis in boiling-water reactors amounts to one molecule per 100 eV of absorbed energy. Such a yield of radiolytic products of water can be expected only when the steam generated in the reactor is discharged at very high rates, but this is not observed under real process conditions.

Under real conditions, the yield of radiolytic products of water decomposition is much lower, since partial recombination of the radiolysis products does occur at the actual steam flowspeeds observed. The yield of radiolytic oxygen in the reactor of the second power unit of the BAES power station amounts to 10 Nl/MW·h, which corresponds to the oxygen concentration of 5-6 mg/kg in the saturated steam downstream of the condenser.

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TABLE 1. Quality Standards for Water and Steam in BAES Power Station Second Power Unit during Operating Period

Parameters standardized	Feedwater	Reactor circulation water	Reactor scavenging water	Saturated steam and superheated steam	Turbine condensate
Total hardness not higher than, μg-equiv/kg	3	15	-	-	3
Alkalinity (according to mixed indicator, ammonia subtracted) not higher than μg · equiv/kg	-	-	50	-	-
Sodium (determined on a flame photometer) not higher than, μg/kg	-	-	-	-	10
Silicic acid (SiO_3^{2-}) not higher than, μg/kg	30	-	1000	20	-
Chlorides (Cl^-) not higher than, μg/kg	-	30*	-	-	-
Iron oxides (Fe), not higher than, μg/kg	-	60	-	-	-
Copper content (Cu) not higher than, μg/kg	5	-	-	-	5
Total content of corrosion products not higher than, μg/kg	-	-	500	-	5
Oxygen content (O_2), μg/kg	10	-	-	-	30
Oils content not higher than, μg/kg	300	-	-	-	-
pH, not lower than	-	8.0	-	-	-

* In case of accident, chlorides content increase to 150 μg/kg in circulation loop water allowable for 20 hours, for every 1000 hours reactor has been in operation.

No further increase in water radiolysis products is observed in the steam generating channels of the reactor of the BAES power station second power unit when nuclear steam superheat is in effect.

Gaseous radiolytic impurities in the coolant (oxygen, hydrogen) enter the turbine entrained in the steam, and then proceed along with the bled steam into the regenerative preheaters, where they are responsible for aggravated corrosion of the steam – water passages.

Taking into account the special features of the corrosion behavior of the structural materials, it is found necessary to maintain weak ammoniated water conditions on the second power unit of BAES. These conditions are characterized by maintenance of a weakly alkaline medium (pH = 8 to 9.5) in the coolant, through the introduction of ammonia solutions. The maximum ammonia concentration is set by the allowable corrosion of the copper base alloys, at 2 mg/kg.

Some misgivings arose in the design stage to the effect that such relatively low ammonia concentrations, which are inadequate for suppressing radiolysis of water, might cause the formation of nitrite and nitrate ions as complete decomposition took place in the radiation field, in combination with oxygen. This might have the effect of driving the pH value below seven. As operating experience demonstrated, however, the rate of formation of nitrates is very low, and these have generally no substantial effect on the pH of the coolant.

Efforts to maintain the pH of the feedwater at the 8-9.5 level had beneficial effect on the pearlitic steels. The condensate – feedwater passages, and the PVD above all, which are made entirely of pearlitic steels, operate at temperatures not above 215-217°C. The dissociation constant of ammonium hydroxide (NH_4OH) at temperatures of 150° and 200°C is still reasonably high ($7.41 \cdot 10^{-6}$ and $3.3 \cdot 10^{-6}$, respectively), so that the pH value at these temperatures differs negligibly from the room-temperature pH values. The 2 mg/kg ammonia content in the water thereby brings about a pH not lower than 9.0, at t = 150° to 200°C.

As mentioned earlier, the channels and fuel-element cladding in the BAES reactor (second power unit) are made of stainless steel. In the evaporation channels, where the boiling takes place, real conditions for concentration of chlorides are brought about, and when radiolytic oxygen is present at the same time this may mean corrosion cracking of the stainless steel. As a consequence, the chlorides content in the reactor circulation loop water is restricted by very stringent regulations.

Table 1 lists the operating norms for quality of coolant in the second power unit of the BAES power station.

In order to maintain the content of the salt impurities and corrosion products in the coolant within tolerable limits, the second power unit of the BAES power station has provisions for cleanup of condensate, and also for purging water from the reactor circulation loop.

TABLE 2. Actual Parameters of Coolant Quality for Second Power Unit of Belyi Yar (BAES) Nuclear Power Station during Normal Operating Period

Parameters determined	Feedwater	Reactor circulating (condensing) water	Reactor blowdown water	Saturated steam	Superheated steam	Turbine condensate
Total hardness, μg · equiv/kg	< 3	< 3	3 ÷ 6	—	—	3
Silicic acid (SiO_3^{2-}), μg/kg	—	—	100 ÷ 300	5 ÷ 15	5 ÷ 15	—
Chlorides (Cl^-), μg/kg	25	25	25	—	—	—
Oxygen, μg/kg	10—15	30	30	(5 ÷ 6)·10^3	(5 ÷ 6)·10^3	40—50
Ammonia, mg/kg	1 ÷ 2,5	0,6 ÷ 1,4	0,6 ÷ 1,4	0,8 ÷ 2	0,8 ÷ 2	1 ÷ 2
pH value	9,2 ÷ 9,5	8 ÷ 9	9 ÷ 9,5	9 ÷ 9,5	9 ÷ 9,5	9 ÷ 9,5
Iron (Fe) oxides, μg/kg	20—60	20—60	30—60	20—30	20—30	10
Copper (Cu), μg/kg	—	—	7 ÷ 30	0,4	—	0,8
Specific activity, Ci/liter	—	—	10^{-5}	—	10^{-7}	—

A mechanical washed-on filter is a sufficiently effective filter for removing corrosion products during the startup period. Specially processed sawdust is used as the filter medium.

The measures taken to seal the turbine condensers against coolant leakage allow for bringing steam traps into service periodically.

The design amount of blowdown for the reactor circulation loop amounts to 1.5% of the throughput. But after startup and adjustment operations have been carried out in steady state, the amount of blowdown was cut back to below 1%. This blowdown was controlled as a function of the chloride content in the water of the reactor circulation loop. A method with a sensitivity of 1 μg/kg is being used at the present time to measure the chloride concentration. The chloride content in the water of the reactor circulation loop under stationary operating conditions is not raised above the standard level (30 μg/kg), and is kept, on the average, at the level of 15 to 13 μg/kg. The amount of continuous blowdown of the loop is set below 1% throughput.

The blowdown water from the reactor in the second power unit is directed to a specialized water treatment facility, where the blowdown water purged from the first and second power units of the BAES power station are processed simultaneously. The identical water conditions maintained in these two power units justify the use of a single facility for treating the blowdown water stream.

Careful deaeration of both the feedwater and the feedwater components is carried out in order to cope with corrosion of the condensate – feedwater passages. Noncondensable gases are drawn off from the PVD and PND vessels. The condensates from the heating steam of these condensers are sent in cascades to the (PVD) deaerators and (PND) condensers, where they are deaerated to an oxygen content of 10 to 15 μg/kg in the water. The oxygen content in the feedwater (downstream from the deaerator) is 10 to 15 μg/kg.

Experimental work on binding the radiolytic oxygen in the PVD heating steam by adding hydrazine hydrate solutions to the steam paid off. In this case the oxygen content in the heating condensate amounted to not more than 0.03 μg/kg.

As a result of all the measures taken with the object of maintaining an optimum set of water treatment conditions, the quality of the coolant on stream at the second power unit of the BAES power station is now characterized by the data appearing in Table 2.

Tables 1 and 2 show clearly that all the parameters of the water treatment process fall practically within the standard range during the normal operating period.

Constant checks are carried out on the state of the surface of pearlitic steels in the second power unit of the BAES power station. Inspection of different process equipment made of pearlitic steel made it clear that equipment components experiencing service under the same conditions as at ordinary fossil-fuel power generating stations (deaerators, feedwater vessels and lines) behave in the same manner as their counterparts in the conventional power stations.

Part of the equipment undergoing heavier duty because of the presence of radiolytic oxygen experiences pitting corrosion to depths of 0.3 to 0.5 mm, in addition to the overall uniform corrosion.

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In view of the relatively short service life of the equipment, we cannot reach any definitive conclusions on the prospects for widespread application of pearlitic steels in the fabrication of PVD equipment for boiling-water reactor power stations.

We can, however, draw the following inferences from our study.

1. Weakly alkaline water conditions brought about through metered introduction of ammonia in amounts of up to 2 mg/liter, and bringing the pH level of the coolant to 9.5, have proved acceptable for structural materials of the types used in the second power unit of the Belyi Yar nuclear power station (BAES).

2. Maximum sealing of the turbine condensers makes it possible to work with steam traps brought into play periodically.

3. Three years of operating experience with the second power unit of the BAES power station have shown that no serious malfunctions occurred in process equipment made of pearlitic steels, during this period. But because of the relatively short service life of the equipment, it would be premature to draw any definitive conclusions as to the applicability of pearlitic steels for process equipment continually in contact with coolant containing milligram concentrations of oxygen (specifically, the vessel and PVD vessel and piping).

PREPARATION FOR STARTUP OF THE A-1 NUCLEAR POWER STATION

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An important problem in the construction of the first Czechoslovak nuclear electric power generating station, the A-1 nuclear power station, consisted in comprehensive testing of the power station equipment and introducing the power station to reliable service. The testing and startup period of the A-1 power station proceeded in four distinct stages.

First Stage. This stage covered comprehensive testing of the process equipment, and was carried out gradually, on separate operating complexes one at a time. The problem was to check out the equipment in nonradioactive environments with the reactor filled with distilled ordinary water. The sets of equipment were accepted by the purchasing plant after complex testing procedures in conformity with the report. The testing program and the list of conditions under which the purchaser will accept the set of equipment are stipulated in the report. The coordination part of the project includes a review of the interaction of operating sets of equipment in complex testing of the A-1 power station and its equipment.

Second Stage. Next followed the preparation for the physical startup and power startup. This stage involved a conclusive checkout of the equipment and is characterized by the transition to active operations, e.g., to fueling of the core, checkout of the loops, etc. The principal operations planned for this stage are: complete monitoring and inspection of leaktightness of the primary loop; checking out the loaded core during short-term thermal tests; rechecking the functioning and quality of the process equipment.

Third Stage. The physical startup of the reactor occurs in this stage. The nuclear characteristics of the reactor operated at minimum power, and with minimum hookup to the other equipment of the A-1 power station, are checked in this stage. During preparations for physical startup, a special program was worked out for scheduling 19 experiments at the following parameters: zero reactor power output, coolant pressure from 1 to 9 atm; air passed through the primary loop, followed by CO_2 in the later phases.

The physical startup program envisaged the following experiments: experiments on attainment of the critical level of heavy water in the reactor completely loaded with fuel elements; calibration of control and scram rods, and determination of the period over which the reactor is brought up from low power, as a function of the moderator level; checkout of the automatic power control system; measurements of neutron distributions.

Since it takes 121 days to complete such an expanded program of experiments, the possibility of shortening the physical startup time in some way was looked into. The outcome of this study was a so-called basic startup program to back up the expanded startup program. Startup times in this program are shortened substantially by including only those experiments which are of paramount importance for safe and efficient operation of the reactor, and also by carrying out several of the experiments simultaneously. The total time required to complete the physical startup according to the basic program is 31 days.

Fourth Stage. This stage covers the power startup of the reactor. A special program was also worked out for this stage, and comprises the general principles of nuclear power station startup procedures envisaged in engineering plans, and flowing from the design solutions of the A-1 power station process equipment.

Calculations of several stationary modes of operation of the A-1 power station were carried out on an electronic computer, in order to work out particular testing procedures and power startup technological

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TABLE 1. Operating Modes during Power Startup

Mode	Fuel element cladding temp., °C	Reactor thermal power, output, MW	Temperature of CO_2 at reactor exit, °C	Electrical power rating of one turbine, MW	Medium-pressure steam			Low-pressure steam		
					flowrate, m/h	pressure, atm	temperature, °C	flowrate, m/h	pressure, atm	temperature, °C
1	250	29	232	—	11,4	5,7	227	4,5	2,1	148
2	300	44	274	—	16,3	6,8	271	4,9	2,1	151
3	350	54	316	—	15,7	6,9	308	5,9	2,1	165
4	300	38	274	—	20,7	6,7	270	6,2	2,1	158
5	300	67	272	—	25,3	6,4	269	7,6	2,1	150
6	300	91	271	5	24,5	15	269	19,8	2,1	174
7	350	150	311	6,5	46,2	16	306	23,8	2,1	172
8	350	213	308	11,6	65,3	15	308	31,6	2,1	171
9	385	250	337	15	79,6	16	328	34,3	2,1	171
10	400	265	350	22	87,0	15	338	32,8	2,1	169
11	400	408	346	35	120	23	335	65,4	2,1	181
12	450	488	386	45	148	28	374	68,6	2,1	184
13	450	403	389	51	178	33	377	89,9	2,1	190
14	500	569	427	53	176	34	413	70,8	2,1	190

operations in detail, while retaining the basic features of the plans. On the basis of analysis of several variants, 14 basic operating modes of the electrical power generating station were proposed, for establishing a tentative sequence to bring the power station up to full power for the first time (see Table 1). The basic concept underlying the startup procedure was to gradually raise the power output of the nuclear power station, while bearing in mind the fact that the reactor is the most important piece of station equipment, and that the decisive process parameter is the temperature of the fuel-element cladding. The entire power startup phase is broken up into three phases, which differ in the process parameters and in the way the nuclear power station is operated.

The first phase of the power startup stage involves operational tests in which the A-1 power station is brought up to 10% of rating, with the CO_2 pressure ≈ 10 atm, and the temperature of the fuel-element cladding rising to 350°C. During this period, production of electric power is not the focus of attention, and the actual power needs of the station itself are satisfied from an extraneous power source. The operation of the nuclear power station is characterized by low process parameters in this period, and this is responsible for the fairly low accuracy of the measuring systems, while eliminating the corrective effects of some controls (e.g., the CO_2 temperature at the reactor exit, the pressure at the entrance to the gas blowers, etc.). At the same time, the servicing personnel and technicians of the power station are confronted with more stringent requirements on operation of a nuclear power station.

The most important operating tests are carried out during the first phase of the power startup, and these are, in particular:

warmup of the reactor and of the primary loop;

startup of the reactor and operation of the reactor at a level ~2.5% of rating;

operation at different stationary modes, and transitions from one mode to another;

taking the nuclear power station off the power grid and hooking it back into the power grid;

de-energizing the power station in response to a scram signal.

Experiments and checkouts of process equipment not carried out in the course of the preceding comprehensive tests were also included.

The second phase of the power startup is characterized by attainment of power and parameters at which the turbines can be switched on. The temperature of the fuel-element cladding is 350°C at this point, and the pressure at the entrance to the gas blowers is 35 atm (with three turbines operating). Experiments involving checkout of process equipment, carried out in the course of the comprehensive testing program, are also included in the second phase.

In the third phase of the power startup, the nuclear power station will be operating at parameters close to ratings (see Table 1, mode No. 12), specifically with the fuel-element cladding temperature at

TABLE 2

Number of experiment	Content of experiment
1	Testing emergency protection system of reactor
2	Testing fuel assembly overload conditions when fuel assembly is transferred from core to cooldown zone, and vice versa
3	Checking operation replacing fuel assembly with the aid of the refueling machine
4	Determining the corrosion rate of avial metal, with the aid of sensors installed in the heavy-water loop
5	Determination of the characteristics of the control-rod system over the power ranges near 2.5, 10, and 50% of maximum reactor output rating
6	Testing the flowrate and steam bypass correction unit
7	Checkout of matching of control-rod system to power controls and to gas flowrate controls
8	Testing the system of transportable ionization chambers in the measurement mode
9	Checkout of matching of transportable ionization chamber system and control-rod system in control mode
10	Measurement of characteristics of controls for process channels
11	Experiments for supplementing performance measurements and confirmation of relationship between performance measurements and experimental measurements
12	Investigation of the distribution of solid particulates and corrosion products in the first loop
13	Investigation of oil content in gas
14	Measurement of stresses in the primary loop, straingage measurements, measurements of plastic deformations
15	Checkout measurements of prototype equipment for heavy-water loop in active service: determination of CO_2 desorption from D_2O at expected pressure and temperatures; measurement of amount of detonating gas, determination of the characteristics of the formation of detonating gas in response to various factors; experimental verification of the effect of the pressure on catalyst stability in contacting process equipment for combustion of detonating gas
16	Determination of the characteristics of some automatic control networks
17	Comprehensive testing of one of the steam generators
18	Mass-scale experiments with fuel elements: post-irradiation inspection measurements; investigation of thermomechanical operating conditions for fuel elements; investigation of temperature fields and heat release in eight experimental assemblies
19	Determination of temperature coefficient of reactivity of the fuel
20	Measurement and processing of dynamic characteristics of distributing equipment
21	Measurement and processing of characteristics of automatic operating modes
22	Determination of effect of damage to fuel elements on amplitude of signal emitted by system monitoring leaks in fuel-element cladding.

450°C and the pressure at the entrance to the gas blowers at 54 atm (with three turbines on stream). The gas blowers operate practically at rating, and the loads on the steam generators and turbines are also close to ratings. The automatic power controls are given a final checkout. The experimental power startup program is an extremely ambitious one, as we see. Moreover, the CO_2 condensation cleanup system will also be tested in the concluding stage.

The power startup modes are determined on the basis of the following points:

gradual rise in the temperature of the fuel-element cladding to the maximum temperature (450°C), and initial rise in the temperature of the fuel elements achieved with lowered coolant pressure;

as a rule, the rise in power is achieved by allowing only one parameter to vary with all the remaining parameters kept constant;

the secondary-loop pressure must be consistently below the primary-loop pressure;

the minimum pressure in the primary loop is determined by the minimum D_2O pressure at the intake of the pumps (5 atm);

the maximum pressure in the loop is determined by the pressure in the cooling chamber, i.e., by the rated pressure.

The program of experiments on the process equipment included the following operations:

final checkout of process functions not investigated up to that point, since these tests are carried out most conveniently while the power startup is actually in progress;

completion of the first tests on separate systems and subassemblies;

completion of research work aimed at definitively determining the capabilities and performance of various pieces of equipment, etc;

adjustments on the various automatic control systems;

carrying out scheduled additional checkouts on equipment, needed as indicated in the first experiments, or as a result of tests involving some hazard.

These experiments are dictated by the special operating conditions of the nuclear power station. The experiments include checkout of scram shutdown performance and scram cooldown of the reactor. At first, the power startup program provided for scram shutdown of the reactor three times with the object of checking the performance of the equipment under those conditions, as well as providing extra checks on the necessary safety requirements for the nuclear power station. Test scram shutdowns were planned for maximum reactor parameters. Later on, the number of scram shutdowns was increased to seven, including one test shutdown in the first phase, another in the second phase, and five shutdowns in the concluding stage of the power startup.

The list of power startup experiments is given in Table 2.

Some changes have to be made in the regular systems in order to carry out some of these experiments. This refers above all to the measuring and control systems, which must ensure the required accuracy in measurement of parameters even when the nuclear power station is functioning off-rating (at low parameters). Some provisional instruments, to be removed and replaced by the regular instruments designed for the rated parameters, are therefore installed in the measuring and control systems prior to commencing the power startup.

A program containing procedures and ways to process the results, a list of recommended changes and additional equipment, and information on the personnel needed to carry out the experiments, was worked out for each of the experiments. The program included coordination rules for maintenance and operation of the process equipment during the startup. The maintenance rules cover the state of the process equipment at the start of each phase, and include instructions for all the important technological operations.

Certain defects in the equipment were detected and eliminated in the course of working out the program. At the present time, several technical problems relating to preparations for the power startup are being solved. The major problems in this area are:

providing for the plant's own power needs from the 110 kV and 220 kV networks (this is termed "combination power supplies");

the problem of ripple due to the action of the gas blowers operating in steady state, and under emergency conditions;

stability of the gate valve acting as check valve at the exit from the gas blowers.

The design period for the power startup is 300 days, broken down as: 51 days for the first phase, 76 days for the second phase, 173 days for the third phase, the remainder for the fourth. The overall time includes 30 days intercalated between the first and second phases, however, as required to replace the measuring instrumentation and make some changes in certain pieces of equipment.

The building of the first Czechoslovak nuclear electric power generating station has been a complex and crucial task. Successful introduction of the A-1 nuclear power station to the nation's power grid requires maximum efforts on the part of a large group of specialists, equipment manufacturers, scientific institutes, design and planning institutes, and of course the workers and staff of the nuclear power station itself. Soviet specialists rendered no small part of the assistance in solving the problems relating to the startup of the power station.

The preceding results demonstrate that all of the technical and engineering problems appearing in the course of preparations for getting the power station on the line can be successfully resolved.

The prevailing conditions give every ground for optimism that the first nuclear power station in the country will be delivering electric power to the nation's power grid on the planned schedule by the end of 1971.

MONITORING GAS-TIGHTNESS OF FUEL ELEMENT CANS IN GAS COOLED REACTORS

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The occurrence of cracks in fuel element cans in gas (CO_2) cooled reactors, fueled with metallic uranium increases the likelihood of the fuel – coolant reaction:

$$U + 2CO_2 \rightarrow UO_2 + 2CO;$$

$$U + 2CO \rightarrow UO_2 + 2C.$$

Oxidation of the fuel promotes enlargement of the cracks and may lead to ignition of the can itself [1]. For this reason, particular importance attaches to development of systems for monitoring the gas-tightness of fuel element cans (FECM system). In the Czechoslovak nuclear power development program, directed mainly towards gas cooled reactors at the present time, considerable attention has been given to the FECM problem [2-4]. This article describes the main results of investigations made in this field of study and, in particular, of certain problems in choosing methods for detecting lack of gas-tightness in fuel elements and in theoretical optimization systems.

Choice of Method for Detection of Lack of Gas-Tightness in Fuel Element Cans and Basis of Fuel Element Can Monitoring (FECM) Systems. The following requirements must be met when developing a FECM system [5]:

guarantee of reliability and of rapid detection of lack of gas-tightness in any fuel element, using the minimum number of detectors;

capability of observing progressive development of failure (particularly in the case of failure through gradual enlargement of a defect);

incorporation of a suitable method for recording results to assist rapid and straightforward diagnosis;

maximum degree of automation in operation of the system;

incorporation of the FECM system into the reactor control and protection system.

The above-mentioned requirements can be met by two methods in gas cooled reactors, viz., by electroprecipitation or by filtration. Both methods are based upon deposition of "aerosol" daughter decomposition products (isotopes of rubidium and of cesium) in the form of gaseous fission fragments and by measurement of the activity of the deposited dispersion phase. Detectors for FECM systems based on the electroprecipitation principle are adopted in almost all nuclear power stations having gas cooled reactors [6]. At the A-1 nuclear power station, the first in Czechoslovakia, it is proposed to adopt a filtration method using rigid or inflexible filter elements. Filtration detectors, having replaceable filtering surfaces have been tested in experimental gas loops.

The cost of the FECM system in a gas cooled reactor may constitute an appreciable fraction of the whole cost of the power station equipment. Therefore, attention was paid when developing the FECM system that all the aforementioned requirements should be met with minimum number of detectors. As the result, the preferred method selects samples of the coolant from grouped power channels but with periodic monitoring of individual channels. On detection of a channel containing a defective fuel element, a so-called "tracking detector" is switched in, which will record the activity of the coolant continuously until such time as the defective fuel element is removed.

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TABLE 1. Comparison of Signal Magnitudes and Sensitivity of Different Detectors

Type of detector	Signal magnitude, impulses/sec · cm^2	δ, mm^2
Electroprecipitator:		
with wire electrode	9 640	10,4
with disk electrode	400	22,8
Filtration detector having replaceable filtering surfaces	10 670	16,7

Having regard to the large amount of data to be processed and to the importance of this data as a source of information regarding the functional state of the reactor, it is desirable to couple the FECM system to a computer in the automatic control and protection system.

Gas chromatography is a promising method for detection of microcracks, through which long-lived fission products might flow, and offers the possibility of selectively determining isotopes of krypton and xenon in the coolant [7].

Detection of lack of gas tightness in fuel elements with defects which develop rapidly can be effected by constant measurement of coolant samples from the first bank of channels using detectors of the same type as that used in the FECM system itself. This additional monitoring is especially important in view of the method adopted, of selecting samples from grouped power channels with periodical monitoring of individual channels. Theoretical analysis indicates that high rapidity of action is also a characteristic feature of monitoring fission products in coolant, based on methods of recording delayed neutrons [7].

The basis of the adopted FECM system ensures a high degree of reliability over a wide spectrum of possible types of defect in fuel element cans and ensures the provision of valuable information as regards the behavior and the general condition of the fuel charge.

Appraisal of the Sensitivity and the Rapidity of Action of Detectors in Fuel Element Can Monitoring (FECM) Systems. Methods of appraising the sensitivity of detectors and the dependence of this sensitivity on various parameters have been proposed in order to decide the functional conditions of the detectors and to optimize the entire basis of the FECM system. By sensitivity, here, and in what follows, is implied the

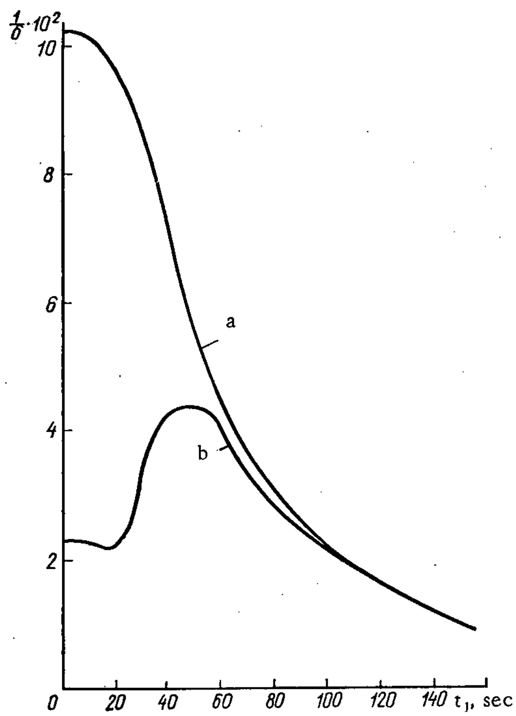

Fig. 1

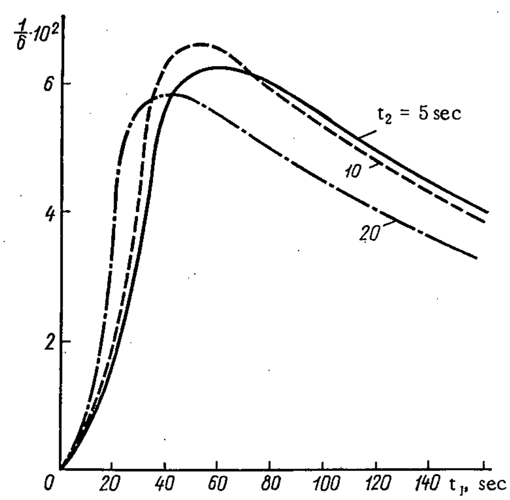

Fig. 2

Fig. 1. Dependence of sensitivity δ of an electroprecipitator with respect to time t_1 of transfer of coolant sample (a) with purging, and (b) without purging of the detector by clean gas.

Fig. 2. Dependence of sensitivity δ of a filtration detector with respect to time t_1 for transfer of the coolant sample for various values of t_2.

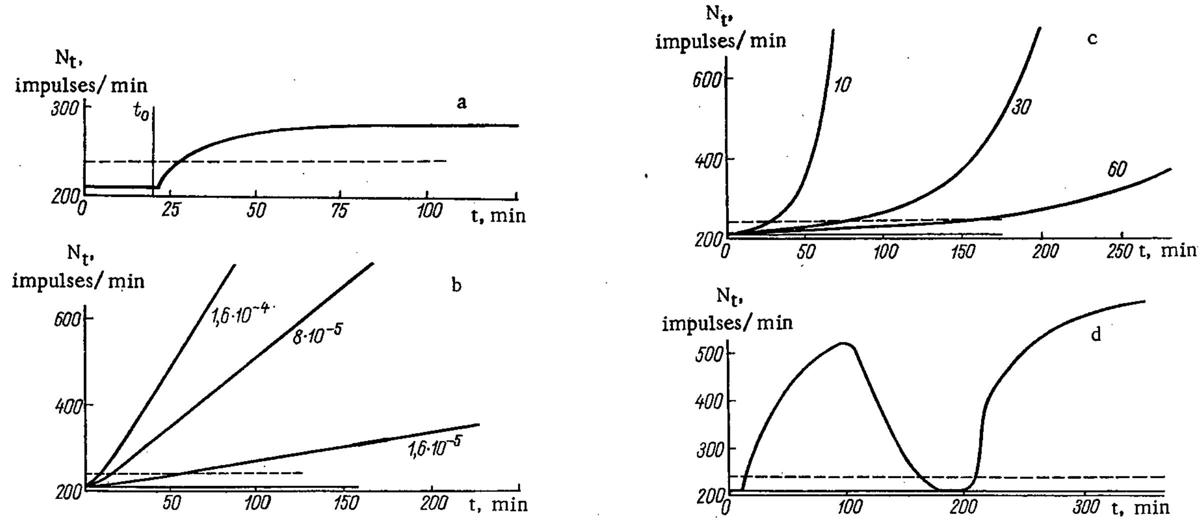

Fig. 3. Time-wise dependence of indication response N_t of FECM detectors in the following cases: a) defects of constant magnitude; b) defects with linear rate of growth (the figure given on the curves are numerical values of r in cm^2/sec); c) defects with exponential rate of growth (the figures on the curves are numerical values of $T_{1/2}$ in minutes); d) defects with occurence of oxides. The dotted lines represent the threshold signal; unbroken horizontal lines represent the background (noise) of the detector.

extent of exposure of the fuel surface which will cause an indication response from the detector equal to the background level. This definition enables rapid appraisal of the effects of the functional and constructional parameters of the detectors and the FECM system upon its sensitivity and assists comparison of different types of detectors.

The reading or indication given by any type of detector can be characterized by the relationship [8]:

$$N_{ijk} = F_i \sum_{j=1}^{n} A_j M_k,$$

where F_i expresses the mechanism of the outflow of fission products; the form of the term A_j depends upon whether the signal is computed for a predetermined defect (A_1) or for the background level (A_2); M_k characterizes the type and the construction of the detector. Each of the terms enumerated may frequently involve a lengthy mathematical expression, particularly complex in the case of a diffusion mechanism of outflow of fission products through microcracks [9].

The sensitivity, δ, according to the definition stated, can be expressed by the relationship

$$\delta = \frac{N_p + P_1 + P_2 + N_{i2k}}{N_{i1k}},$$

where N_p is that part of the background noise of the detector adopted which is caused by external radiation; P_1 and P_2 are the fractions of the background noise of the detector caused, respectively, by surface contamination of the active zone by uranium or by activation of the coolant together with its content of contaminants.

Figure 1 shows the results of computations of the magnitude of δ with respect to the time t_1 taken for transfer of the coolant sample for electroprecipitation using wire electrodes. The dependence of the magnitude of δ with respect to the time t_1 for transfer of samples to the cleaning filter for various times, t_2, for disintegration of the sample between the filter cleaning the gas flow and the filter-collector for isotopes of rubidium and cesium, is shown in Fig. 2. The magnitudes of signals for a given area of fuel, and the magnitudes of δ for optimum parameters of different detectors and for the proposed operational parameters of the nuclear power station are given in Table 1.

Theoretical determination of the minimum detectable magnitude of defect, based on the root-mean-square fluctuation of the background level of the detector, demands more detailed study of the FECM system as a whole and, possibly, appraisal of fluctuations of the main parameters of the system and of the reactor.

If a FECM system is chosen mainly from consideration of rapidity of action, the important thing in detecting lack of gas tightness is the threshold signal strength, S, in the absence of any defect. For computation of the magnitude S, the following relationship can be used [10]:

$$S = N_f + 0.1N_f + \frac{5}{\sqrt{2\tau}}\sqrt{N_f},$$

where N_f is the background level of the detector; τ is the time constant of the integrator.

Rapidity of action depends, also, on the rate of growth of the defect and upon the corresponding time-wise relationship of the response indications of the detector. In appraising the FECM system adopted for the A-1 nuclear power station, attempts were made to calculate the time-wise dependence of the response indications of the FECM detectors for the following types of defects in fuel cans [11]: 1) defects of constant size; 2) defects with linear rate of growth; 3) defects with exponential rate of growth; 4) exposure of a definite area of fuel surface causing oxidation of the exposed surface, spalling-off of the oxide, and repeated exposure of the fuel.

The first three types of defects can be characterized in the following manner:

1. Defects of constant size:

$$M_h = K_0(1 - e^{-\lambda_i' t}),$$

where K_0 is a constant; λ_i' is the disintegration constant of the daughter products of the fission fragments (rubidium, cesium).

2. Defects with linear rate of growth:

$$M_h = K_0 r\left[t + \frac{1}{\lambda_i'}(e^{-\lambda_i' t} - 1)\right],$$

where r is the rate of growth of the defect (cm^2/sec).

3. Defects with exponential rate of growth:

$$M_h = K_0\frac{\lambda_i'}{b+\lambda_i'}(e^{bt} - e^{-\lambda_i' t}),$$

where $b = \ln 2/(T_{1/2})$; $T_{1/2}$ is the period for "half-value growth" of the defect.

The mathematical expression for the time-wise dependence of the signal response in the case of a defect of the fourth type is rather too lengthy for reproduction within the limits of the present article.

Computed time-wise dependence of the response indications of FECM detectors, corresponding to the given types of defects are shown graphically in Figs. 3a, b, c, and d.

The types of defects considered do not include all possible types of defect and their description may not fit the actual case completely. Comparison of the results from theoretical analysis with data obtained in actual service of FECM systems will facilitate the processing and the interpretation of measured results.

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EXPERIENCE IN THE BUILDING AND STARTUP OF THE BOR-60 REACTOR

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The development of the BOR-60 began in late 1963 under the scientific supervision of the Power Physics Institute (FÉI). Construction work on the building to house the reactor installation at the Atomic Reactor Scientific-Research Institute (NIIAR) was begun in mid 1965, and the reactor reached its power level on December 27, 1969.

While producing a high specific power and high sodium coolant exit temperature, the reactor can be used to test fuel elements, and structural materials for the BN-350 and BN-600 fast reactor nuclear power stations now being built, over a wide range of testing conditions, and can serve the same purpose for the future higher-output nuclear power stations using fast breeder reactors. The 40-60 MW power level makes it possible to test different types of equipment under operating conditions, including steam generators, pumps, heat exchangers, shutoff valves, control valves, etc.

The reactor characteristics, and a description of the basic reactor devices, can be found elsewhere [1-4]. This paper deals with problems involved in the installation of the reactor and in the completion of startup and adjustment operations.

Experimental Groundwork for Equipment Designs

All of the basic equipment assemblies were tested, as a rule, on simulating models and prototypes. These include:

top parts of seals for rotating reactor plugs;

models of pressure-vessel pipe connections;

model of pressure chamber in combination with models of fuel assemblies and flow-throttling devices;

models of refueling channel and gating devices;

drives and moving parts of control rod assemblies;

system for bringing rotating plugs home to specified coordinates;

system for monitoring sodium temperature at exit from fuel assemblies.

The eutectic of tin and bismuth displayed excellent wetting with the blades of the seal for the rotating plugs, which provided helium pressuretightness in both hot and cold states. Additional heat-stable rubber sealant failed to provide a stable sound seal, and the unsuccessful profile was altered.

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Models of pipe connections for the reactor pressure vessel were subjected to strength tests under axial loading and a bending moment. Stresses attaining their maximum at the site where the pipe connection is joined to the pressure vessel damp out completely at distances roughly equal to the maximum radius of the pipe connection.

The safety margin for anticipated stresses was not less than two. Later on, when piping is tightened up in the cold state for welding operations, the actual stresses were found to be below the levels anticipated.

Hydraulic tests and temperature tests of a model of an exit pipe connection disclosed that the temperature gradients at the site where thermal shields were placed were below those predicted, at a level of 0.15°C/mm, while the temperature gradient reached a level of 1.4°C/mm outside the thermal shield. Additional shields, longer ones, were installed at the exit pipe connections as a results of the tests, during the assembling process.

Stress measurements taken at different points of the pipe connection made it possible to estimate the allowable number of cycles of rapid changes in the sodium temperature at the exit from the reactor (2000 cycles at 580°C). Strain gages were bonded to the pressure vessel and to the pipe connections of the reactor in order to observe the operating conditions experienced by the pipe connections under full-scale conditions.

The temperature conditions under which the rotating plugs operated were tested on a model, revealing that the temperature in the hydraulic seals reaches 81-85°C even when the water cooling system breaks down, while reaching 68°C on the plug cover. Temperatures are lowered by 35-40°C when air cooling is used.

Simulated pressure chambers and fuel assemblies were tested in order to check the ease of assembly of the core, and the stresses generated when rod bundles are inserted into or withdrawn from the core.

Building and Assembly

It took 4.5 years to complete the building of the reactor. This period can be broken down in the following stages.

1. From late 1965 to mid 1967. The bulk of the work in laying the protective concrete shields and precast structural concrete, wall panels, installation and rigging of metal structures and metal lining of rooms housing the primary loop, was completed. Heavy shielded doors were installed in those rooms where access was to be limited as construction proceeded further on the building. Hoisting and lifting equipment was set up. Construction work on the building housing the turbine and the ancillary equipment for the steam loop was begun later.

2. From late 1967 to mid 1968. Auxiliary systems and service systems for the main building, not connected with the sodium stream, and not subject to special purity requirements in the assembly stage, were set up, and load-bearing structures and the reactor shielding were rigged into place. Primary-loop rooms were prepared for the installation and erection of equipment and the main piping. A room provided with rigging stands was prepared for preliminary work with the pressure vessel and other reactor components.

3. From late 1968 to mid 1969. Erection of the principal and auxiliary sodium loops, reactor components, electrical power system, monitoring and automatic control systems, was completed. Critical experiments were run in the absence of coolant in order to secure more exact data on some of the physical parameters. The "dry" physical startup of the reactor was carried out in December, 1968.

4. From August through December, 1969. Final adjustments were made on the principal and auxiliary sodium systems, the operating conditions for electrical heating were tested, sodium was admitted into the auxiliary systems and purified in cold traps, the principal-loops were filled with sodium, the performance of the pumps and other equipment was checked out, the necessary characteristics were measured, fuel assemblies were loaded in, critical experiments were conducted on the sodium stream, and the power startup was completed.

The design of the reactor facilitated fabrication of the pressure vessel and rotating plugs separately. No serious problems arose in matching the mating components properly, since there was still one more component, the reactor "cage," to intervene between the pressure vessel and the rotating plugs. The pressure vessel and the "cage" were fabricated first, and went through tests together.

In assembling the reactor, it became necessary to align the axes of the control rod mechanisms as precisely as possible with the corresponding axes of the discharge header axes, to hold the specified clearances, and to check the ease of assembly of the core. The congested tangle of equipment on the reactor cover caused difficulties in getting the equipment into place, and necessitated readjustments and refitting in some instances.

It was not possible to avert serious welding deformation in the vicinity of the exit pipe branches when the guard rings were installed in the reactor cage at the site of fabrication. The dimensional deviations were found to be excessive, since further difficulties might eventually hamper the free movement of the rotating plugs. Because of a lack of time to remedy that defect, the cage was turned over for assembly and installed in the reactor pressure vessel to facilitate "dry" critical experiments without sodium present. After these operations had been completed, the reactor was dismantled and the dimensional deviations were eliminated.

After the equipment had been assembled on the reactor cover and the operation of the mechanisms had been checked out in the cold state, an argon environment was set up in the reactor and in the primary loop, for the purpose of preventing oxidation of the tin–bismuth alloy as the sodium was being admitted into the loop. Permanent insulation of the alloy was achieved by application of silicone fluid. With that, the preparation of the reactor for final adjustment operations in the hot state was completed.

Welded joints were subjected to tests for strength, helium leak tests, γ-ray nondestructive testing, and dye-penetrant crack detection tests. After welding operations had been completed, the piping butt-weld joints on the high-temperature service lines from the reactor to the sodium – sodium heat exchangers were subjected to austenization at temperatures to 1100°C. The flaws detected in the welded joints in the primary loop in two instances were eliminated with comparative ease.

Completion of the rigging, installation, and assembly, testing and run-in of the secondary loop with the air heat exchanger took place simultaneously with the filling of the primary loop with sodium, and the run-in and critical experiments for the primary loop. This made it possible to bring the reactor power up to 5 MW in late December, 1969, with the air heat exchanger working.

Startup and Adjustment Operations

The primary loop was tested for vacuumtightness and dryness. Helium leak tests were conducted first, and all leaks detected were eliminated. Steel simulators of working bundles and shield bundles were loaded in. The alloy in the hydraulic seals of the rotating plugs was in a frozen state. The pumps were provided with bearing-stand seals for gas shutoff. No oil was admitted into the pumps.

The primary loop was dried out by evacuating it and simultaneously applying the electrical heating system. The temperature of the primary loop was raised in 50°C steps. The maximum loop temperature in the tests was 300°C. The reactor pressure vessel was warmed up by pumping hot air (~300°C) over the shell of the pressure vessel.

The loop was filled with argon several times during the drying process. Pure argon with an oxygen content of $3 \cdot 10^{-3}$% and moisture content 0.1 mg/liter was used in this operation.

The loop was completely evacuated, in all of its complexity, for a full five-day period. The limiting vacuum attained in the loop was $2 \cdot 10^{-1}$ mm Hg, with inleakage of $8 \cdot 10^{-2}$ mm Hg per hour.

By November 26, 1969 the loop was ready for filling with sodium. The main loop piping was at temperatures of 200-240°C, the auxiliary piping was at temperatures of 200-300°C, the intercooler vessel was at 150-250°C, the reactor pressure vessel was at 180-200°C. Thermocouples installed in the reactor gave readings of 170-220°C.

Preparation of the Sodium. Sodium from the manufacturing plant was delivered in receptacles one cubic meter in volume, under a layer of argon. From 25 to 1000 g paraffin (used at the manufacturing plant as a protective medium to keep the sodium insulated from attack by air) was driven off under vacuum from each shipping container, as the sodium in the receptacle was heated to 200-250°C. The paraffin continued to be driven off until paraffin vapors ceased to precipitate out in the air-cooled trap installed at the intake of the vacuum pump, and this process took three hours. The sodium was forced out from the shipping receptacle by the argon, through a mechanical gauze filter, into an intermediate tank 6 m^3 in capacity, where the sodium was allowed to settle for some time, after which it was forced up from the bottom into a 35 m^3

capacity receiver tank. All the tanks were dried out, before sodium was admitted into them, by evacuation backed up by simultaneous electrical heating.

The intermediate and receiving tanks containing sodium were evacuated via the vapor trap, in order to monitor the amount of paraffin that might have gained access to them. When the intermediate tank filled with the first batch of sodium was evacuated, some traces of paraffin were detected in the vapor trap. The tank was evacuated over a 19 hour period, until paraffin ceased to show up. No paraffin was detected in the trap when the intermediate tank with a second batch of sodium was evacuated, and the same holds for the receiving tank. Sodium from the intermediate tank never got down to the tank bottom.

The sodium in the receiving tank was purified by circulating it through a cold trap. The content of oxides in the sodium was lowered so thoroughly, as a result of tenfold pumping of the volume of sodium through the trap, that the temperature at which the plug indicator stopped fell from 220°C to 120°C. The sodium was transferred from the receiving tank into the primary-loop overflow tank via a mesh filter, and was filtered further by circulation through the standard primary-loop cold trap. Chemical analysis of samples of sodium taken from the overflow tank after the primary loop had been filled showed the following content of principal impurities (%): $4.5 \cdot 10^{-3}$ carbon, $6 \cdot 10^{-4}$ hydrogen, $1.5 \cdot 10^{-3}$ nitrogen, $6 \cdot 10^{-3}$ calcium, $2 \cdot 10^{-2}$ potassium. It should be emphasized that the content of these impurities in the sodium flowing through the primary loop underwent no change as the reactor continued in operation.

Sodium was prepared for the secondary loop in a similar manner. The initial oxides content in the secondary-loop overflow tank corresponded to the temperature at which the indicator became clogged, 195°C. After eight volumes of sodium had been pumped through the cold trap, the clogging temperature of the indicator dropped to 110°C.

The primary loop was filled with sodium at a sodium temperature of 250°C. The loop was first filled with pure argon. The argon was pumped out of the overflow tank by an electromagnetic pump. The filling operation was carried out through the reactor vessel drain channels with the sodium transferred into the loop via raised filters. After the pressure vessel had been completely filled, the filters were relowered, and the loop was filled up via the appropriate drain channels. As circulation began all the sodium from the loop entered the reactor after passing through filters installed, in place of the topworks of the Du-200 globe valves, on the upstream piping of the reactor process equipment.

The loop-filling operation proceeded smoothly, with no mishaps, the sodium pumped by the electromagnetic pumps was monitored by a flowmeter, by the overflow tank overflow gage, and also by changes in the readings of thermocouples in the electrical heating arrangement and readings of the straingaging system.

The only delay resulted from a malfunction of the sodium fill level indicator in one of the intercoolers when the sodium level in the reactor pressure vessel reached its maximum, and this malfunction seems to have been due to the improper indication of the site where the air relief filter was placed. (The indication mark was too high.) The air relief valve had to be cut off from the primary-loop gas system in order to fill the heat exchanger with sodium, and the sodium level in the heat exchanger was raised by venting gas to the atmosphere via a sampling globe valve provided for the purpose.

Straingage measurements taken while the loop was being filled with sodium revealed that stresses reached a level of 300-400 kg/cm^2 in the piping, 600-800 kg/cm^2 in the zone of the lower pipe connection on the vessel, and 2500-3000 kg/cm^2 at some points on the body of the intercooler.

Purification of Primary Loop and Run-In of Pumps. Inasmuch as purging of the sodium loops by water was ruled out, the loop had to be cleansed of any contaminants left in the wake of the installation operations by circulating scavenging sodium through the loop via the mechanical mesh filters installed in place of the topworks of the delivery-end globe valves.

Circulation of sodium through the loop showed that the mechanical pumps with lower hydrostatic bearing performed excellently. But when sodium was circulated through one of the filaments of the primary loop, severe hydraulic impacts occurred along the loop because of chatter of the check valve on the second filament of the loop. The frequencies and amplitude of these oscillations in the loop increased with pump shaft rpm. No loop oscillations were observed when two pumps were operated in parallel.

A clogging temperature of 180°C was obtained when the first measurement of oxides content in the primary loop was taken. The clogging temperature of the plug indicator was found to be 110°C after the oxides cold trap had been in operation for 58 hours.

The loop was cleaned out by circulation through the mesh filters for seven full days, with maximum flowrates as high as 320 m^3/h through each filament. After that the mesh filters were taken out of the loop. Inspection of the filters revealed no foreign inclusions of any kind present. The check valve responsible for the chatter in the loop was removed at the same time, and was replaced by a check valve with a heavy-duty swing. After the check valve had been replaced, sodium circulation through the loop was resumed, using a pump with a lighter check valve, and no perceptible chatter was reported. The further run-in of the loop took place at a sodium flowrate of 320 m^3/h and temperature of 200-300°C. The pump characteristics and pump idling behavior were measured during that time, and emergency interlocks were checked out. By December 5, 1969, the loop was ready for loading working bundles into the reactor, and "wet" physical startup with coolant present began.

Filling and Run-In of Secondary Loop. As in the case of the primary loop, the secondary loop was filled after prolonged drying of the loop by evacuation combined with simultaneous heating to 200-250°C. The secondary-loop piping was baked out by electrical heating, and the air heat exchanger was baked out by gas burners installed in each of the four sections.

The limiting vacuum attained in the loop was $6 \cdot 10^{-1}$ mm Hg, and inleakage was $5 \cdot 10^{-2}$ mm Hg per hour.

The two filaments of the secondary loop and the air heat exchanger were filled up with sodium simultaneously, but prior to that the freedom of passage of all the drain lines was checked out. The sodium level in the banks of pumps reached the upper critical level at the start of the buffer tank level gage readings. Gas flow through the buffer tank had to be shut off before circulation through the secondary loop was begun, therefore, and the sodium was partially transferred from the pump tanks to the buffer tank. As a consequence, the buffer tank was replaced by a tank of larger volume set at a lower mark.

Sodium circulation through the two filaments of the loop was started in parallel through the air heat exchanger, and proceeded smoothly without incident. No loop vibration was observed. The oxides content in the loop, measured immediately after circulation commenced, corresponded to the indicator clogging temperature of 170°C.

Sodium was circulated at flowrates up to 220 m^3/h through mesh filters (0.25 × 0.25 mesh) which replaced the topworks of the globe valves on each filament of the loop. All the filters were removed after six full days of circulation of the sodium stream.

The core was loaded at a faster pace with the aid of a simplified accessory, a rod suspended from a hook on a hoisting crane. Loading was done through the loading hole in the small rotating plug, without a gating device.

A hose supplying a continuous flow of argon was lowered into the hole to keep air from getting in.

The simulating bundles withdrawn from the reactor were placed in a box containing soda. The loading rod was washed free of sodium residues in an alcohol – water solution. The working bundles were heated to 150-180°C in an electric furnace before being placed inside the reactor. The sodium temperature was 180°C throughout the reactor loading process. The oxides content in the sodium rose toward the end of the loading operation, up to the clogging temperature 150°C.

After the physical startup of the reactor had been completed, the reactor was brought up to 5 MW power in December, 1969, at a sodium temperature of 350°C, with heat removed by the air heat exchanger.

In March, 1970, the reactor power was brought up to 20 MW at sodium temperatures to 400°C. By August 15, the reactor had generated 47,000 MW of heat power.

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THE EFFICIENCY OF FUEL UTILIZATION IN POWER-GENERATING FAST REACTORS

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The efficiency of fuel utilization at an atomic power station is defined primarily by the cost of the fuel cycle to generate 1 kWh of electrical energy. The criterion used is the fuel component of the unit costs. In addition to the monetary indicators, it is also helpful to make use of natural indicators of the fuel cycle, such as the uranium or plutonium requirement of a particular type of reactor, the power doubling time, the amount of work involved in the manufacture and chemical processing of the fuel elements, and the magnitude of the separation by the enriching industry.

Each of these indicators characterizes one particular aspect of the fuel cycle, even though they are all interrelated. They are free from some of the vagueness and lack of definition that is characteristic of monetary indicators.

Fuel Consumption in a Breeder Reactor System

An important aim in any consideration of the variables for the development of a nuclear power system is to minimize the rate of consumption of fissionable materials. The consumption of fissionable material at any instant of time is the sum of the amount expended to install new reactors and the amount expended on fuel make-up for reactors installed earlier. For the sake of simplicity, we shall assume that new increments of nuclear power are put into operation at an exponentially increasing rate, i.e.,

$$\frac{dW(t)}{dt} = Ae^{\omega_0 t}, \text{ and hence } Wt = \frac{A}{\omega_0}(e^{\omega_0 t} - 1),$$

TABLE 1. Comparative Characteristics of Fast Reactors with Different Types of Fuel

Indicator		Type of fuel			
		oxide (10% burnup, reproduction factor = 1.33)	carbide (10% burnup, reproduction factor = 1.43)	metal (2.5% burnup reproduction factor = 1.63)	metal (5% burnup, reproduction factor = 1.61)
Charge of plutonium in the cycle, tons	T_r = 1 yr	3.3	3.5	5.8	4.1
	T_r = 0.5 yr	2.7	3.0	4.0	3.2
Doubling time, yr	T_r = 1 yr	10	8.2	9.2	6.8
	T_r = 0.5 yr	8.2	6.9	6.4	5.2
Plutonium consumption for development with a specified rate and reprocessing time, tons	T_{sp} = 2 yr, T_r = 1 yr	2.6	2.6	4.6	2.9
	T_{sp} = 3 yr, T_r = 1 yr	2.3	2.2	3.9	2.3
	T_{sp} = 5 yr, T_r = 1 yr	1.7	1.4	2.7	0.6
	T_{sp} = 5 yr, T_r = 0.5 yr	1.1	0.8	0.9	0.2

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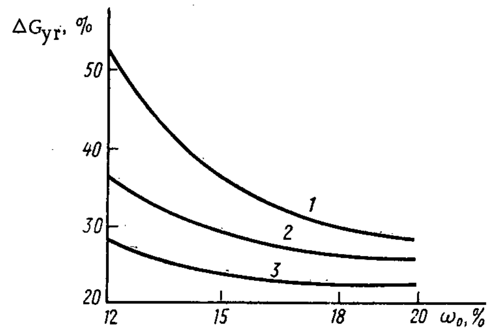

Fig. 1. Influence of the change from $T_t = 1$ yr to $T_t = 0.5$ yr on savings in annual plutonium consumption at various values: 1) RF = 1.5; 2) RF = 1.4; 3) RF = 1.3.

TABLE 2. Relative Change in Plutonium Consumption as φ Changes, %

Value of ω	$\varphi = 1,0$	$\varphi = 0,8$	$\varphi = 0,6$	$\varphi = 0,4$
0,12	−10,5	0	+25,0	+34,0
0,20	~0	0	~0	~0

and

$$\omega(t) = \frac{1}{W} \cdot \frac{dW}{dt} = \frac{\omega_0}{1 - e^{-\omega_0 t}}.$$

This law of power increase (a high relative rate of increase at the beginning of development, decreasing with time to a value ω_0) describes the qualitative properties of the dynamics of a developing nuclear power industry far better than the exponential law of power increase that is sometimes used, $W(t) = W_0 e^{\omega t}$. The value of ω_0 must be taken equal to the asymptotic rate of growth and the entire power-generating system. The latter will scarcely exceed 7-8% per year. The annual consumption of fissionable material for such a development of a system of fast reactors using plutonium can be found from the approximate expression

$$G_{yr}(t) = G_{cy} \frac{dW(t)}{dt} - G_{cy} \omega_c W(t) = W(t) G_{cy} [\omega(t) - \omega_c], \tag{1}$$

where G_{cy} is the specific quantity of plutonium in the fuel cycle, expressed in tons/million kW (electrical); ω_c is the characteristic rate of growth of fast-reactor power that corresponds to a doubling time of $T_2 = 0.693/\omega_c$.

The lower the specific quantity of plutonium in the fuel cycle and the higher the reproduction factor, the lower the plutonium consumption will be. The increase in the power density of the fuel as the size of the active zone is reduced does reduce the specific charge of the fuel cycle in most cases, but at the same time it markedly reduces the reproduction factor. The specific charge per cycle may be represented approximately in the form

$$G_{cy} \simeq G_a + G_{yr} T_r = G_a + \frac{1000\beta}{\bar{B} \cdot \eta} 365 \varphi Z_0 T_r \tag{2}$$

where G_a is the charge of the active zone, in kg of Pu/million kW (electrical); G_{yr} is the annual overload, in kg of Pu/million kW (electrical) · year; $\bar{B}$ is the average burnup fraction, expressed in MW · days/ton; β is the reactor power fraction that is assignable to the active zone; η is the efficiency; φ is the load factor; Z_0 is the initial concentration of the plutonium (kg/ton of fuel); T_r is the time of the external fuel cycle, expressed in years.

The value of G_a is inversely proportional to the power density of the fuel and to the efficiency. Consequently it is advantageous to increase the power density of the fuel, within reasonable limits, bearing in mind the engineering considerations, the unavoidable reduction of the internal reproduction factor, and some reduction in the total reproduction factor, as well as the increase in the amount of fissionable material in the external fuel cycle. The latter is related to the increase in the initial concentration of plutonium in the fuel (Z_0) with smaller active-zone volume values.

An efficiency value of 40-43%, which would be obtainable at steam parameters of 130-160 atm absolute and 500-540°C, is completely acceptable. Increasing the efficiency will also reduce the specific quantity of fuel in the external cycle. However, more powerful factors come into play here: the burnup fraction and the time required for the external fuel cycle.

Increasing the burnup fraction has the advantage of reducing fuel distribution costs; at the same time, however, the amount of fuel in the external part of the cycle also decreases, as can be seen from formula (2). This effect becomes stronger as the initial value of the burnup fraction decreases and as the duration of the external cycle increases. For a given burnup fraction, shortening the external cycle will reduce the amount of fuel in external cycle, and the smaller the given burnup fraction is, the stronger this effect will be.

TABLE 3. Characteristics of Uranium Reactors

Characteristic	LWR	HWR	FOB
Specific charge of uranium, tons of natural uranium/million kW (electrical)	400	130	650
Annual consumption of uranium for recharging (operation at 100% of power), tons of natural uranium/million kW (electrical) year	180	110	185
Annual production of fissionable plutonium (operation at 100% of power), kg of Pu/million kW (electrical)·yr	220	400	800

The doubling time also depends on the specific quantity of fissionable material in the fuel cycle:

$$T_2 \simeq \frac{G_{cy}}{(RF - 1)\varphi}. \tag{3}$$

Figure 1 shows how reducing the duration of the external fuel cycle from 1 yr to 0.5 yr affects the annual plutonium consumption of a fast reactor (using plutonium oxide) with an average burnup fraction of 70,000 MW·days/ton for different values of the specified rate of growth of the power-generating system. The ordinate axis shows the annual savings in plutonium consumption, while the abscissa axis shows the specified rate of annual growth of nuclear power generation. It is clear from the figure that reducing the length of the external cycle brings substantial savings in plutonium consumption and that the relative role of the reproduction factor in the plutonium savings diminishes as the specified growth rate of nuclear power generation increases.

In selecting the reactor variant which is optimal from the point of view of plutonium consumption, we must, of course, keep in mind the fact that at different stages in the development of nuclear power generation the requirements imposed on such reactors will be somewhat different. During the initial period of development, when the rate at which automically generated electrical power is assumed to be increasing most rapidly, preference should be given to reactors with a minimum amount of fissionable material in the fuel cycle, since the value of the reproduction factor for a high specified rate of growth of power generation will be only of secondary importance. Later, when the specified growth rate is smaller and consequently the amount of plutonium used for putting reactors into operation is more nearly equal to the excess amount of plutonium produced, we find that the influence of the reproduction factor on the annual plutonium consumption increases considerably and may become decisive (where the specific quantity of fuel per cycle and the value of the reproduction factor are bounded between reasonable limits of variation).

This fact can be illustrated by an example comparing reactors which use different types of fuel: oxide, carbide, and metal.

Assume that all three reactors have the same volume (5000 liters), the same composition (40% fuel, 40% sodium, and 20% steel), and the same power [1000 MW (electrical)]. For the oxide and the carbide the maximum allowable burnup is taken to be 10%, while for the metal fuel (a U – Pu – 10% Zr alloy) it takes on values of 2.5% and 5%. The fuel density is taken to be 8 g/cm^3 for the oxide, 10.5 g/cm^3 for the carbide, and 11 g/cm^3 for the metal (70% of the alloy density, which is 16.5 g/cm^3). The calculations were carried out for pure Pu^{239} with no admixture of higher isotopes.

The results of these comparisons are shown in Table 1. As can be seen, for high values of the specified development rate there is no need to use metal or carbide fuel, even if the metal fuel will reach a burnup value of 5%. The use of metal fuel can be advantageous only if the specified rates of development are sufficiently low and the burnup value is approximately 5%. The full economic advantage of using metal fuel is not yet evident at this stage, since the cost of electrical energy will be determined not only by fuel costs but also by the cost of preparation and distribution, as well as the capital component. Obviously, first priority must be given to introducing oxide fuel, and second priority to carbide fuel.

A change in the above-mentioned reactor parameters can also be achieved by a suitable choice of the reactor design (the choice of the active-zone geometry and the percentage of the volume represented by fuel coolant, and structural materials) and the conditions of operation of the atomic power station. For more economical fuel consumption, it is desirable to have fast reactors operating with high load factors (φ), although during the initial period of development this is of no great importance, since in this case the

TABLE 4. Characteristics of Plutonium Reactors

Characteristics	FOB	FCB
Specific charge of fissionable plutonium, tons of plutonium/million kW (electrical)	2,2	2,0
RF	1,40	1,55

reproduction factor plays only a minor role, and the quantity of fuel in the cycle diminishes with decreasing φ. Thus, at high rates of growth the fuel consumption will remain almost unchanged (Table 2).

Fuel Consumption in a Mixed Reactor System

For a realistic model of the development of nuclear power engineering, we should assume that the reactors used during the initial stage of development are thermal reactors, since those are the reactors most thoroughly understood today. Fast-neutron breeder reactors will begin to develop somewhat later, when a sufficient amount of plutonium has been accumulated. As converter reactors, we may use thermal reactors of various kinds, as well as fast-neutron uranium reactors. The latter produces somewhat more plutonium than the commonly used light-water reactors and may be converted to breeder operation when free plutonium is present. A disadvantage of such reactors is that their specific charge of uranium is greater than in the case of thermal reactors.

We give below the results of calculations of natural-uranium requirements for the development of nuclear-power generation to 600 million kW after 30 years [1]. In this connection we shall investigate how the uranium consumption depends on the type of uranium reactors used, the time when fast reactors are first introduced, the replacement of oxide fuel in fast breeder reactors by carbide fuel, the reduction of the length of the external cycle of fast reactors, and the conditions of operation of the atomic power stations.

In all the variants the development of nuclear power generation begins with light-water thermal reactors (LWR), and in the first variant it continues until the end of the period under consideration. In all the other variants, except the tenth, fast plutonium reactors with oxide fuel (fast oxide breeders – FOB), using plutonium that has been processed in thermal reactors, are first put into operation ten years after the beginning of the development process. In the tenth variant this happens after 15 years instead of ten.

In addition to fast plutonium reactors, other types of reactors are constructed as needed: either light-water thermal reactors (LWR), heavy-water reactors (HWR), or fast uranium reactors using oxide fuel (FOU). In the last case the construction of thermal reactors stops ten years after the start of the development process (in the tenth variant 15 years after). In variants 11 and 12, 15 years after the beginning of the development process the oxide-fuel plutonium breeder reactors are replaced by carbide-fuel breeder reactors (FCB).

We give below some characteristics of all the reactors used in the calculation (Tables 3 and 4). The characteristics approximately correspond to the average values of the data given in the literature [2-4]; for the uranium reactors the characteristics are given in terms of equivalent quantities of natural uranium, assuming a 0.25% concentration of U^{235} in the tailings.

The length of the external fuel cycle is assumed to be one year for thermal reactors, while for fast reactors this length is reduced to 0.5 year in two variants (the ninth and twelfth) 15 years after the start of development. The load factor is 0.8 except where otherwise indicated (variants 3 and 4).

The specified time for doubling the power T_0 is taken to be 2.5 years for the first decade, 4.0 years for the second decade, and 8 years for the third decade.

The results of the calculations are shown in Table 5. Here, in addition to the total consumption of natural uranium over 30 years, we also give the consumption of uranium necessary after this period in order to make all the thermal reactors operate until the end of their service life (30 years).

The use of fast reactors makes it possible to reduce the consumption of natural uranium by a factor of almost 3 in comparison with LWR reactors alone. The delay in the introduction of fast reactors causes an increase of more than 20% in the consumption of uranium (variants 6 and 10). An important role is played by the use of FOU fast reactors instead of LWR during the second and third decades. Because more plutonium is produced in this case, plutonium reactors are preferable to LWR reactors and at a certain point in time they will not only become self-supporting but also make it possible to replace uranium fast reactors with plutonium fast reactors; the reason for this is that a surplus of plutonium will be produced because the rate of development gradually decreases.

TABLE 5. Consumption of Natural Uranium after 30 Years as Nuclear Power Generation Develops to a Total of 600 Million kW

Variant	Total consumption after 30 years, thousands of tons of uranium	Commitment of thermal reactors, thousands of tons of uranium
1 LWR	885	1720
2 LWR + FOB	730	1100
3 LWR + FOB (φ = 0.7 for all reactors after the first ten years of development)	685	1000
4 LWR + FOB (φ = 0.7 only for the LWR after the first ten years of development)	670	870
5 LWR + HWR + FOB	400	375
6 LWR + FOU + FOB	500	33
7 " " " (change in reproduction factor = +0.1)	450	33
8 " " " (change in reproduction factor = −0.1)	555	33
9 " " " (duration of external cycle for fast reactors is T_f = 0.5 yr for the last 15 years)	385	33
10 LWR + FOU + FOB (five-year delay in the introduction of fast reactors)	640	200
11 LWR + FOU + FOB + FCB (FCB is introduced instead of FOB after 15 years)	360	33
12 LWR + FOU + FOB + FCB (T_r = 0.5 yr when the introduction of FCB is begun)	300	33

A considerable saving in natural uranium can be made by using heavy-water thermal reactors (TWR) instead of LWR reactors during the second and third decades; in variant 5 the consumption is 45% less than in variant 2.

The reduction of φ occurring over a limited range of indicator values in the variants with thermal reactors not only does not increase the uranium consumption but, on the contrary, results in reduced uranium consumption because the consumption for the LWR reactors decreases. From a comparison of variants 3 and 4 we can see that if φ is reduced to 0.7 at the same time in thermal reactors and fast reactors, then the increase in consumption caused by the deterioration in reproductive properties of fast reactors in this case is small, amounting to only 2% of what it is when φ is reduced only in thermal reactors. A change of ±0.1 in the reproduction factor produces relatively small changes in the total uranium consumption (within ±10%).

For the case in which nuclear power generation develops with the use of fast reactors, the total uranium requirement for the 30 year period is considerably less than for the case in which only thermal reactors are used. In the future, when the characteristics of fast reactors are improved and the time required for the external fuel cycle is reduced, this advantage of fast reactors will become even more evident.

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SELECTION OF PARAMETERS FOR A LARGE FAST REACTOR

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The minimum cost of electric power at which the required doubling time is provided should be the optimization criterion in the selection of the basic parameters of a nuclear power station using fast reactors. Taking this position, such power station characteristics as mean heat release rate, core flattening, breeding factor, thermal efficiency, etc., are considered as parameters which are necessary for optimization [1, 2].

The total station power, average heating, and maximum coolant temperature determine both the thermal engineering, as well as the physical, characteristics of the installation. For their choice, overall optimization of the station is necessary.

Experience in designing and constructing nuclear power stations with BN-350 and BN-600 reactors furnishes a basis for assuming that an individual power station of 1000-1500 MW (e) is now completely reasonable [1]. An installation of such size makes it possible to reduce specific capital costs and the fuel component of power cost in comparison with the BN-350 and BN-600.

As noted in [2], it is economically inadvisable to turn to the highest parameters of the thermodynamic cycle in constructing fast power reactors. Considering this, the final choice of cycle parameters must be made by accepting a compromise solution. On the one hand, an economic advantage appears with a reduction in temperature because of the increase in reliability and the greater breeding factor; on the other hand, the efficiency is reduced because of the degradation of steam parameters.

Power release rate and core flattening, blanket thickness, and methods for equalization of energy deposition determine mainly the fuel cycle costs and therefore can be selected independently of thermodynamic cycle parameters.

TABLE 1. Classification of Competing Factors Determining Optimum Heating

I. Increase in heating permits	What cost item the factor affects and how	II. Reduction in heat permits	What cost item the factor affects and how
1) An increase in total fraction of fuel (reduction in coolant fraction), which leads to:		1) An increase in station efficiency;	Reduces FC and CC
a) an increase in the diameter of fuel elements and a decrease in their total number;	Reduces cost of fuel manufacture and processing (FC)	2) An increase in reliability (because of less danger of thermal shock);	Reduces CC
b) an increase in the breeding factor;	Reduces FC	3) A reduction in the effect of steel and fuel expansion at maximum permissible temperatures	Reduces FC and CC
2) Reduction in coolant pumping costs;	Reduces operating cost (OC)		
3) Decrease in number of feed-water heaters;	Reduces capital costs (CC)		
4) The manufacture of individual parts such as steam generator economizer and piping from lower quality steels;	Reduces CC		
5) An increase in the overall reliability of the station	Reduces CC and FC		

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For that reason the following discussion can be divided arbitrarily into two parts.

Selection of Thermodynamic Parameters

The basic parameters are the sodium exit temperature and the average heating in the reactor. At the present time, there is still no data for a final determination of the optimum values of these parameters because there is not yet sufficient experience of the operating reliability of the basic equipment. However, specific tendencies can be verified by means of calculations.

We consider the problem of the selection of the optimum coolant heating for fixed exit temperature including superheat factors at constant reactor power. In Table 1, the factors tending to increase and decrease the optimum heating are divided into two groups.

In the numerical computations that were performed, factors 1) and 2) of the first group of factors and factor 1) of the second group were taken into account. The most important input data for a power station having a $UO_2 + PuO_2$ core as fuel is given in Table 2.

The efficiency was calculated for the thermodynamic cycle of a nuclear power station with a 500 MW turbine and a steam pressure of 160-130 atm. The change in costs of coolant pumping is taken into account in the value for the station efficiency.

The main results of the calculation are given in Table 3.

The results point to the following. Fuel costs are markedly reduced because of the increase in BF and d_{fuel} despite the resulting decrease in efficiency and heat release. Furthermore, the effect of the increase in BF and fuel element diameter, which leads to a reduction in FC (allowing for the decrease in efficiency and Q), turns out to outweigh the effect of the reduction in thermal economy, which leads to an increase in CC. In summing these competing effects, the total cost of electric power in a given range of $\Delta \bar{T}$ decreases with an increase in heating.

Similar calculations were also made for a hypothetical version with a steam pressure of 90 atm. For such a steam pressure and atmospheric deaerating of water, it turned out reactor heating considerably above 250°C can be realized. The optimal heating was 260-270°C (for minimal total cost of electric power).

The quantitative results obtained confirm earlier conclusions as to the basic nature of the optimum values for the thermal parameters of a nuclear power station with a fast reactor. We also consider it necessary to draw attention to the fact that far from all the factors tending to increase the optimum heating (reduction of average temperature in the reactor) are taken into account in the numerical calculations.

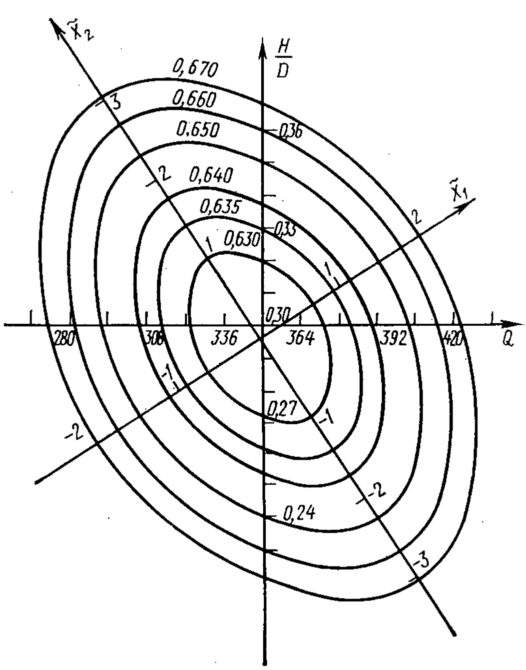

Fig. 1. Dependence of FC on heat release Q [kW/liter] and flattening β = H/D.

Selection of Reactor Core and Shielding Parameters

Average Heat Release (Q) and Flattening (β). The average heat release and flattening $\beta = H/D$ (H and D are respectively the height and diameter of the core) for a given thermal power of an installation determine the dimensions of the reactor tank and other quantities which mainly affect the fuel component (FC) of the calculated cost. There is an optimum in the dependence of FC on heat release and flattening. This can be shown by calculations using the methods presented in [3].

TABLE 2. Input Data for Calculation

Parameter	Value
Thermal power W_T, MW	3750
Reactor exit temperature T_{ex}, °C	530
Specific power Q, kW/liter	500
Flattening	0.4
Average burnup, %	10

TABLE 3. Computed Results

Characteristic	T_{ex} = 530° C		
	$\Delta\bar{T}$ = 200° C	$\Delta\bar{T}$ = 230° C	$\Delta\bar{T}$ = 245° C
Station efficiency (net), %	40,4	39,7	39,4
Fuel fraction	0,34	0,39	0,41
Breeding factor (BF)	1,44	1,48	1,49
Diameter and thickness of fuel cladding, mm	5,8 × 0,3	6,3 × 0,3	6,5 × 0,3
Heat release rate Q, MW(t)/kg	1,61	1,52	1,49
Total power cost, %, including:	100	98,0	97,5
capital and operating components (CC + OC), %	100	102,0	103,0
fuel component (FC), %	100	88,0	84,0
Fraction of FC in total costs, %	29,0	26,0	25,0

TABLE 4. Blanket Parameter Values

Parameter	Oxide shield		Metallic shield	
	350 kW/liter	500 kW/liter	350 kW/liter	500 kW/liter
$\Delta_{e.b.}$, m	0,380	0,395	0,255	0,265
$\Delta_{l.b.}$, m	0,360	0,400	0,250	0,285
$T_{l.b.}$, yr	2,15	1,82	2,45	2,20
FC_{sh}, kop.$\cdot 10^{-1}$/kW·h	−0,452	−0,620	−0,460	−0,608

For a reactor with plutonium oxide fuel, the ranges $200 \leq Q$[kW/liter] ≤ 800 and $0.1 \leq \beta \leq 1$ were considered. The dependence of FC on Q and β was obtained, as expressed by the following approximate equation for W = 5000 MW:

$$\text{FC [kopeck} \cdot 10^{-1}/\text{kW} \cdot \text{h}] = 0.6255 + 1.35 \cdot 10^{-3} Q + 1.91 \cdot 10^{-3}\beta + 4.85 \cdot 10^{-3} Q \cdot \beta + 9.188 \cdot 10^{-3} Q^2 + 6.138\beta^2,$$

which is the equation for an elliptic paraboloid. Figure 1 shows sections of this paraboloid for FC = 0.063-0.067 kop/ kW·h. It is clear that flattening and average power release can be considerably varied without leading to a marked rise in FC. The optimum values of Q and β are in the following ranges: Q = 300-500 kW/liter, $\beta = 0.3$-0.4. The effect of thermal power on the shape of the optimum is insignificant although the absolute value of the FC is a minimum for higher powers. These conclusions point to the possible ranges of the mean release and flattening for a large reactor. Other conditions being equal, high intensity is more favorable since the dimensions of the reactor tank are then minimal.

Blanket Parameters. Calculations indicate that the core characteristics of a large fast reactor can be considered independent of the blanket thickness and material.

The existence of an optimum thickness (for minimum FC) of the lateral (Δ l.b.) and end (Δ e.b.) blankets arises from the fact that the breeding factor increases with an increase in blanket thickness. At the same time, the amount of work in processing and preparing blanket units increases. For minimum FC, the optimum average plutonium residence time in a lateral blanket ($T_{l.b.}$) is determined by two competing factors: a larger loading of plutonium which leads to a reduction of this time, and a decrease in expenditure on preparation and processing which leads to an increase in the time.

Theoretical studies were made for a 2500 MW (t) reactor with plutonium oxide fuel having an average heat release of 350-500 kW/liter. The optimum thickness of metallic and oxide blankets was considered as well as the optimal residence time of the secondary plutonium in the lateral blanket. It was assumed that reloading for one-, two-, and three-zone lateral blankets occurred without rearrangement of blanket units from one zone to another. The maximum possible optimum values for blanket thicknesses were determined (for arbitrary values of the cost indexes); they varied from 0.4 to 0.5 m depending on the type of blanket material and the specific power (350-500 kW/liter). The optimum values of the blanket parameters are given in Table 4.

From the values for FC_b, a metal blanket is nearly as good as an oxide blanket assuming the specific cost of their preparation is identical.

Studies of one-zone, two-zone, and three-zone reloading of the lateral blanket showed that the optimum thickness is independent of the reloading method as indicated in Table 5.

The optimum plutonium residence time in the lateral blanket is nearly completely independent of the number of zones considering the fact that the optimum thickness of a lateral blanket in the multizone version is greater than in a single zone. The use of multizone versions of lateral blankets offers no marked economic advantages over one-zone blankets. The point is that a decrease in the average Pu residence time in internal layers of the blanket (by way of a decrease in the loading in it) leads to a simultaneous shortening

TABLE 5. Dependence of Lateral Blanket Parameters on Reloading Method

Parameter	Metal shield (Q=500 kW/liter)			Oxide shield (Q=500 kW/liter)		
	one zone	two zone	three zone	one zone	two zone	three zone
Δ_{lb}, m	0,285	0,300	0,305	0,400	0,410	0,415
T_{lb}, yr	2,2	2,16	2,15	1,82	1,80	1,79
FC_{sh}, kop $\cdot 10^{-1}$/kW·h	−0,608	−0,617	−0,619	−0,620	−0,627	−0,630

of the operating period, which increases the processing and preparation costs. The use of zonal loading of a lateral blanket may only be necessary where there exist technical limitations on the reactor residence time of blanket units.

Methods for Equalization of the Energy Deposition Field. The production in the reactor core of a stable and uniform energy deposition field is an important means for improving the economic indexes of large fast reactors [4]. The methods used for energy deposition equalization play an important part in the determination of the properties of a fast reactor both as a power device and as a breeder of nuclear fuel. Equalization can be accomplished in two ways: by varying the enrichment of the fissioning isotope and by varying the volumetric fraction of fuel with constant enrichment of the fissioning isotope. The first method is called the method of equalization by enrichment, and the second, the method of equalization by composition [4].

Structurally, the method of energy deposition equalization by composition can be achieved by varying: a) fuel element spacing, b) fuel element diameter, c) fuel density, d) fuel density by filling the cavity with an inert diluent. In contrast to the method of equalization by enrichment, the energy-deposition field is practically unchanged during an operating period with equalization by composition because the breeding factors of the zones are very much the same.

Table 6 presents computed results for the characteristics of reactors with oxide and metallic uranium fuel, in which equalization of energy deposition is achieved by composition (number of fuel elements in an assembly) and by enrichment.

For the average burnup values considered, Table 6 indicates that the breeding factor, fuel element diameter, and average volumetric fuel fraction are higher for the method of equalization by enrichment. It also provides a better economic index with respect to the FC magnitude.

We shall consider briefly the results of calculations which show the relative effectiveness of various methods of equalization based on the variation of the shape of energy distribution and excess reactivity during an operating period and on variation of the magnitude of the average burnup in spent fuel. The

TABLE 6. Calculated Characteristics of Reactors with Oxide and Metallic Fuel and Energy Deposition Equalization by Composition and Enrichment

Parameter	Oxide-fuel reactor		Metal-fuel reactor	
	composition	enrichment	composition	enrichment
Electric power, MW	1500	1500	1500	1500
Thermal power, MW	4000	4000	4000	4000
Average core heat release rate, kW/liter	500	500	500	500
Average coolant heating, °C	270	270	270	270
Average coolant exit temperature, T_{ex}, °C	530	530	530	530
Core flattening	0,3	0,3	0,3	0,3
Average burnup, %	~5,8	~5,8	~2,9	~2,9
Maximum burnup, %	10	10	5	5
Fuel element diameter d_{fuel}, mm	6	6,5	5,2	5,3
Average fuel fraction in core (ε_F)	0,441	0,532	0,363	0,437
Core conversion coefficient (CCC)	0,60	0,67	0,81	0,89
Reactor conversion coefficient (RCC)	1,00	1,05	1,16	1,23
Critical mass (G_{cr}), ton	3,73	3,98	4,79	5,32
Fuel component of estimated costs (FC), kop/kW·h	0,128	0,110	0,202	0,192

TABLE 7. Characteristics of Shaped Reactors

Basic charactertistic	Enrichment shaping			Composition shaping								
				fuel element diameter			fuel element spacing			fuel density		
$\varepsilon_F^{II}/\varepsilon_F^{II}$	1,3			1,6			1,6			1,4		
CCC	0,82			0,81			0,80			0,81		
B^{max} (%)	5	10	15	5	10	15	5	10	15	5	10	15
$\frac{\Delta K_{ef}}{K_{ef}}$ (%)	−1,6	−3,0	−4,3	−2,0	−3,8	−5,4	−2,0	−3,8	−5,4	−2,5	−4,4	−6,2
$\frac{\Delta K_r}{K_r}$ (%)	−4,30	−0,40	+3,50	−0,42	−0,74	−1,10	−0,56	−0,9	−1,2	+0,07	+0,07	−0,68
$\overline{B}$ (%)	2,97	6,25	9,10	3,00	5,80	8,70	3,00	5,85	8,80	3,20	6,40	9,75
Operating period (yr)	0,375	0,796	1,200	0,375	0,752	1,150	0,425	0,775	1,200	0,400	0,845	1,290

calculations were performed for two-zone cylindrical reactors with a core volume of 5700 liters, an average heat release Q = 500 kW/liter, and plutonium oxide fuel. Core flattening was 0.3. It was assumed that the shaped reactors being compared had identical average volumetric fuel fractions $\overline{\varepsilon}_F$ in their cores. This approximation makes it possible to obtain an answer to our question without complicating the calculations.

For the method of equalization by composition, the ratio of the volumetric fuel fraction in the second zone to that in the first zone was varied over the range $\varepsilon_F^{II}/\varepsilon_F^{I} = 1.3$–$1.7$. The greatest average burnup (for a given maximum) is achieved when $\varepsilon_F^{II}/\varepsilon_F^{I} = 1.6$, which corresponds to the maximum core operating period. This is obtained for all methods of equalization except for density shaping, where the optimal value in this regard is $\varepsilon_F^{II}/\varepsilon_F^{I} \cong 1.4$. For enrichment equalization, the minimum variability factor is achieved for a ratio $\rho^{II}/\rho^{I} = 1.3$ of the concentrations of the fissioning isotope in the second zone to that in the first zone. These conclusions are independent of the values of the average energy release and maximum burnup.

The calculations demonstrated that all methods of equalization by composition are equally effective from the viewpoint of obtaining a stable energy distribution during the operating period. In shaping by fuel enrichment, significant equalization of energy distribution occurs, as can be seen from Table 7, in accordance with the relative change in the coefficient of variability $\Delta K_r/K_r$.

All the shaping methods discussed provide approximately the same average fuel burnup over an operating period although some preference may be given to density shaping. The change in reactivity during an operating period is a minimum for equalization by enrichment, which has the highest internal breeding factor.

It should be pointed out that the value of the CCC for metallic fuel may exceed unity with equalization by enrichment. (Thus for the reactor presented in Table 6, CCC = 1, 2 for enrichment equalization in the breeder mode.) In this case, equalization by composition may be more preferable.

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THE BRG-30 EXPERIMENTAL GAS-COOLED FAST POWER REACTOR WITH DISSOCIATING COOLANT

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Extensive development of the nuclear power industry will be possible only on the basis of fast reactors allowing drawing of all nuclear fuel reserves into the fuel cycle, and facilitating the organization of breeding of fissionable uranium isotopes. The irradiation level of the core has to be increased to the 400-800 kW/ liter range, and coolants which do not soften the neutron spectrum in the reactor, have to be used, in order to attain the required doubling times (6 to 8 years) corresponding to the growth rates of the power industry. Liquid-metal coolants satisfy the necessary requirements on the whole, but the use of liquid-metal coolants in high-output nuclear power stations means a more complicated flowsheet and additional capital investment to handle the problems of incompatibility between sodium and water, induced activity, the need for careful cleanup of impurities, the high melting point, etc. This explains the intense research and development work underway in the USSR and in other countries [1, 2] on gas-cooled fast reactors using heat conversion arrangements which are simpler than the sodium variant. Gas coolants improve the physical characteristics of reactors, and may even lower the cost of power station equipment and the operating costs of nuclear power stations. But reliance on gas coolants awaits the solution of some intricate problems in maintaining the required amount of heat to be extracted from the core, emergency cooldown, and leakproofing of the loop at elevated pressures.

The Nuclear Power Institute of the Academy of Sciences of the Belorussian SSR [IYaÉ AN BSSR], working in collaboration with several organizations and institutes, has ventured onto a new direction in nuclear power studies and development, specifically the use of dissociating systems as coolants and working fluids for process streams in nuclear power plants. A program of research, already completed, and some design developments underway, indicate [3] that the use of dissociating nitrogen tetroxide (N_2O_4), which exhibits some positive physicochemical and heat-transfer properties, such as: low boiling point and low heat of vaporization, high vapor-phase density, radiation-thermal stability, low activation, etc. [4], will make it possible to build nuclear power stations with a simple single-loop arrangement based on a gas – liquid cycle and a gas-cooled fast reactor.

The heat transfer coefficients are increased through highly effective thermal conduction and high specific heat, accountable to the heat of the chemical reactions and the concentration-dependent heat transfer in a nonisothermal mixture, which makes it possible to attain the required level of heat extraction from the core at acceptable pressures (130-160 atm) and at acceptable gas flowspeeds (30-50 m/sec), while markedly improving the weight and size picture of the heat-exchange equipment. Turbines working on N_2O_4 require one-fifth to one-fourth the amount of metal in their design as steam turbines do. At temperatures of 520-540°C and pressures of 130-170 atm, the efficiency of a nuclear power station with N_2O_4 as working fluid is greater than in similar simple cycles based on CO_2, H_2O, He, etc.

The work already done in this area demonstrates some essential improvements (as much as 20 to 30%) in engineering cost figures for nuclear power stations with fast gas-cooled reactors using dissociating nitrogen tetroxide, when compared to nuclear power stations using sodium coolant.

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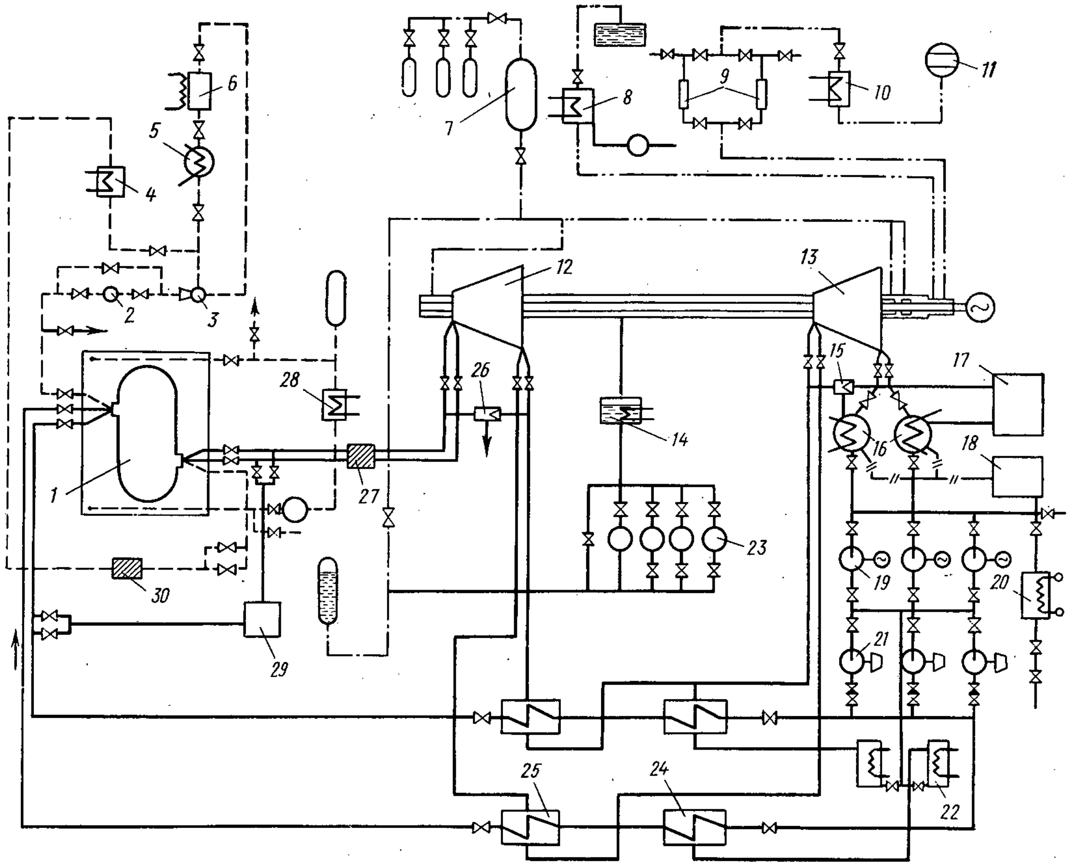

Fig. 1. BRG-30 reactor facility basic flowsheet: 1) reactor; 2) cooldown system gas blower; 3, 11) ejectors; 4, 8) coolers; 5, 10, 16) condensers; 6) fill tank; 7) tanks for turbine lubrication system; 9) adsorber bed; 12, 13) high-pressure and low-pressure parts of turbine, respectively; 14) cooling tank; 15, 26) BROU; 17) coolant cleanup system; 18) vacuum system; 19, 21) low-pressure and high-pressure tanks, respectively, in main loop; 20) make-up system tank; 22) cooler for partial modes; 23) pump for bearing lubricant system; 24) reheater; 25) steam generator; 27, 30) mechanical cleanup filters; 28) condensor for biological shielding cooling system; 29) fired boiler.

An experimental power facility incorporating a gas-cooled fast reactor which uses dissociating coolant (the BRG-30 fast reactor facility) is being developed for service-life tests on fuel elements and reactor systems, for working out the technology of this new coolant, for cleanup and emergency cooldown systems, and in order to accumulate operating experience with the basic subassemblies and subsystems of the power loop, and also to solve problems arising in connection with the development and improvement of recommendations on the design and operation of high-output nuclear power stations utilizing this dissociating coolant.

In assigning the physical and heat-transfer parameters of the reactor, and the characteristics of the thermodynamic cycle, in terms of temperature and pressure, it was assumed that the BRG-30 facility must reproduce the operating conditions of similar high-output reactors rated in the 1000-2000 MW (e) range, in terms of core burnup (450-600 kW) at pressures 130-150 atm, and with the temperature of fuel-element cladding at 720-740°C.

A gas – liquid cycle with intermediate regeneration and intermediate bleed features downstream of the high-pressure turbines was selected for the BRG-30 facility. The reason for this was the relatively low thermal power output of the reactor and the correspondingly modest volume flowrates of coolant. Intermediate bleed with coolant condensed in a condenser – reheater was introduced in order to obtain an acceptable height of blading for the high-pressure turbine, and to facilitate single-pass reactor core design. The basic characteristics of the BRG-30 are listed below:

Thermal power output of reactor	30 MW
Electrical output of generator	11 MW
Coolant flowrate:	
through reactor	62 kg/ sec
through low-pressure turbine and condenser	38.2 kg/ sec
Thermal power output of:	
regenerator	$6.02 \cdot 10^7$ kcal/ h
condenser – reheater	$0.84 \cdot 10^7$ kcal/ h
condenser	$1.75 \cdot 10^7$ kcal/ h

In consideration of the problems to be solved through developing the BRG-30 reactor facility, the efficiency obtained (above 30%) is fully adequate to the purpose.

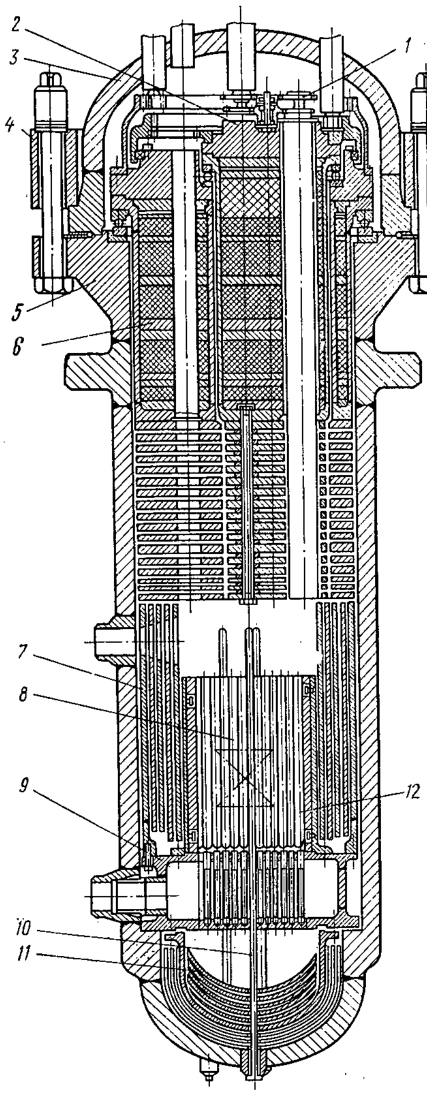

Fig. 2. Longitudinal section through BRG-30 reactor: 1) refueling mechanism; 2) small rotating plug; 3) reactor cover; 4) bolted flange; 5) reactor pressure vessel; 6) large rotating plug; 7) side shield of pressure vessel; 8) fuel assembly; 9) header; 10) control rod assembly; 11) bottom shield; 12) shield assembly.

In view of the great difference in loop pressures, as well as the need for simplicity in equipment fabrication, and the need for improved conditions for preventive maintenance, repair work, and replacements, a circuit variant of equipment arrangement was adopted for the purpose of testing out the various subsystems. The heat-exchange equipment was set up on two autonomous subloops of the main loop. The BROU units were connected up in parallel to the turbines. This makes it possible to operate the reactor with one of the subloops not functioning, or with the turbogenerator shut down. The basic flowsheet of the BRG-30 facility is shown in Fig. 1. The coolant is circulated by the low-pressure pumps 19 and high-pressure pumps 21, whose working fluid is liquid nitrogen tetroxide. Downstream of the high-pressure pumps, the coolant is fed into two parallel subloops, where it is heated in the condenser – reheaters 24, and is superheated in the regenerators 25, and enters the reactor 1 in gaseous form at the temperature 402°C. The gas emerging from the reactor passes through the high-pressure turbine 12 to enter the regenerator, and from there a part of the stream from each subloop is supplied to the low-pressure turbine 13, and from there to the condensers 16. Approximately 38% of the coolant is collected in the reheaters, where it is condensed, and enters the medium-pressure header between the low-pressure and high-pressure pumps.

Reactor

A longitudinal cross section through the reactor appears in Fig. 2. The reactor vessel is cylindrical, while the vessel cover and bottom closure are elliptically domed structures made of 48TS steel clad on their inner surfaces with Kh18N9 steel. The inner diameter (of the cylindrical portion of the vessel) is 1400 mm, the wall thickness 125 mm. The cylindrical part features two entrance and two exit pipe connections for coolant (with inner diameter 150 mm). The coolant flows through the reactor from top to bottom. The top of the reactor pressure vessel features one large and one small rotating plug (6, 2), which, together with the refueling system, facilitate reloading of fuel assemblies and discharge of fuel assemblies, from the core to the spent-fuel cooling area, and from the reactor. The drives for the rotating plugs and refueling mechanism are located on the reactor cover 3.

The core and the side reflector consist of hexagonal cells forming a dense packing. These cells are accommodated by their shanks in the collector 9. The bottom shield cage is coupled to the bottom end of the collector, and the shell with push rods and the side shield of the reactor pressure vessel rest on the top of the collector.

The reflector material is steel 400 mm thick. The side shield of the reactor pressure vessel is also made of steel. The shield under the reactor cover is a layer of steel 480 mm thick, and the rotating plugs consisting of alternating layers of steel and graphite, with a total thickness of 1300 mm, also serve as top shielding.

The basic characteristics of the reactor core are:

Fuel-element diameter	6.9 mm
Fuel core diameter	6.0 mm
Thickness of fuel-element cladding	0.4 mm
Pitch of fuel elements in triangular lattice	7.6 mm
Number of fuel elements in assembly	37
Fuel assembly width across flats	50 mm
Spacing of fuel assemblies	51 mm
Thickness of fuel-assembly cladding	1.0 mm
Core height	450 mm
Number of fuel assemblies:	
at start of campaign	56
at end of campaign	68
Volume composition of core:	
fuel	46.5%
steel	21%
coolant	31%
helium sublayer in fuel elements	1.5%
Uranium loading (90% enrichment)	151 kg
Depth of burnup	10%
Campaign	430 days
Unevenness factor:	
at start of campaign	1.15/1.12
at end of campaign	1.14/1.10
Peak neutron flux	$2.1 \cdot 10^{15}$ neutrons/cm$^2\cdot$sec
Peak fast flux (E > 0.5 MeV)	$1.2 \cdot 10^{15}$ neutrons/cm$^2\cdot$sec
Number of control rods (in reflector)	2
Number of rods in set	5
Number of scram protection rods	2
Specific burnup at start of campaign	502 kW/liter
Peak fuel-element cladding temperature with superheat factors taken into account	727°C
Peak temperature at center of fuel	2300°C
Average flowspeed of coolant:	
at center of core	27.1 m/sec
on periphery of core	16.2 m/sec
Average thermal loading of fuel elements	$1.215 \cdot 10^6$ kcal/m$^2\cdot$h

The designs of the fuel elements and fuel assemblies are similar to the designs adopted for those in the BOR-60 reactor. The fuel elements are hermetically sealed, in cladding of ÉI-847 steel. Each assembly accommodates 37 fuel elements spaced by spirally wound elliptical wire. The steel reflector with a pressure expansion compensating volume, within which excess pressure is established in the fabrication of the fuel elements in order to partially balance off the external pressure exerted by the coolant, is welded to the top of the cladding. The fuel, in the form of pellets of sintered uranium dioxide 90% enriched with U^{235}, is placed in the active part of the 450 mm high fuel elements.

The average burnup of the core of this experimental reactor is set at 500 kW/liter. When heat exchange is intensified by applying an artificially fashioned rough surface to the cladding of the fuel elements, etc., the burnup level of the core can be raised to 700-800 kW/liter without making any substantial changes in the cycle parameters. In this case the heat transfer coefficient at the exit from the central fuel elements is raised from $6 \cdot 10^3$ to $17 \cdot 10^3$ kcal/ $m^2 \cdot h$ while pressure losses in the channel are moderate (3.0 to 3.5 kg/ cm^2).

Primary Loop Equipment

The turbine is a single-shaft axial-flow integrally cast body with horizontal split, once-through flow passages for the high-pressure and low-pressure streams, and opposed flow of gas in order to reduce axial stress levels; the high-pressure part has seven stages, and the low-pressure part has eleven stages. The operating speed of the turbine (3000 rpm) makes it possible to connect the turbine output shaft directly to a 12 MW batch-manufactured generator. The axial thrust bearing and the radial bearings are hydrostatic bearings working on liquid nitrogen tetroxide. A contact seal and a bearing-stand seal are installed on the output end of the shaft, in addition to labyrinth seals. Liquid nitrogen tetroxide is supplied to lubricate the bearings, or clean them out, and cooling is handled by a special lubricating system (see Fig. 1, arrows 7, 14, 23). A special system (8-11 in Fig. 1) handling compression, purification, and dehumidification of air is provided to expedite the work of the shaft-end seal.

Kh18N10TL steel is used for the turbine casing, ÉI-612 steel is used for the turbine runner, rotary blading, and nozzle blading, and Kh18N10T steel, chromium-plated ZKh13 steel, and AG-1500 carbon graphite are used for the parts in the seal assembly.

Taking into account the purpose of the experimental facility, the low volume flowrates of gas, and the fact that shaft rpm is well below optimum, we can infer that turbine operating costs are relatively modest: 0.81 at 6430 kW for the high-pressure part of the turbine, and 0.83 at 4500 kW for the low-pressure part of the turbine.

The main circulating pumps are centrifugal-flow, multistage machines connected directly to their electrical drives. The low-pressure pumps and high-pressure pumps are similar in design. The bearings are hydrodynamic, and lubricated by liquid nitrogen tetroxide. The runners are shrouded, with milled blades and the forward runner disk is welded on. Pump characteristics are listed below:

	Low-pressure	High-pressure
Throughput, kg/sec	19.1	31
Number of stages	5	7
Pressure:		
at entrance, kg_{force}/cm^2	1.4	60
at exit, kg_{force}/cm^2	60.5	147
Shaft rpm	2950	2950
Pump efficiency, %	52	53
Pump power intake, kW	150	386

The casing and the parts on the flow path of the pumps are made of Kh18N10T and Kh18N10TL steels, while the rubbing parts of the pump rotor bearings and thrust journal are made of 9Kh18 steel with enhanced surface hardness, and siliconized SG graphite. In the shaft-end seal, AG-1500 carbon graphite and ZKh13 steel with chromium-plated exposed surface are used.

Regenerator and Condenser – Reheater. The heat exchangers are of modular type, counterflow, smooth-tubed, tubeside flow of high-pressure coolant, shellside flow of low-pressure coolant. The heat exchangers are distinguished by the number of sections included in parallel by the length of the modules. The modular vessel is U-shaped, and made of tubing 83 × 4 mm. The vessel accommodates 19 tubes, sized 10 × 1 mm, spaced 15 mm apart in a triangular lattice.

The total area of the surface available for heat transfer in the two regenerators is 2500 m^2, and 604 m^2 in the two reheaters. Each regenerator comes in 57 sections, the reheaters in 27 sections. Two identical modules are connected in series in each section. The identical design of the modules of the reheater and regenerator, and the use of bare tubes, simplify the fabrication technology, but the total heating surface area increases as a result. The use of modules of optimum size and type, with finned tubes, makes it possible to cut the weight and size of the heat exchangers by several factors while increasing the gas flow-speed.

The condensers are of conventional design, with a horizontal bundle of round bare tubes, and four runs of piping for cooling water. The water is cooled in cooling towers to the rated temperature of +20°C.

One condenser is installed on each subloop. The tube bundle consists of 4864 tubes sized 21 × 1 mm with a triangular pitch of 32 mm. The total surface area available for heat transfer in the condenser is 1481 m^2.

Kh18N10T steel is used in the heat exchange equipment and in the piping.

The positive qualities of dissociating N_2O_4 make it possible to solve the reactor emergency cooldown problem rather simply, in a three-stage operation. Cutoff of the primary loop is followed by short-term blowdown of the reactor with scavenging gas from a pressure cylinder blown through by means of the ejector 3, and with subsequent condensation of the gas in the condenser 4. After the gas blower 2 is started, the gas is moved by forced circulation with cooling to the cooler 4. After residual heat release has been diminished and the temperature of the reactor process equipment has been lowered in accordance with the specified program, the system is brought to long-term cooldown conditions with natural circulation of the liquid nitrogen tetroxide.

Figure 1 (17) shows one of the possible arrangements for purifying the coolant to get rid of liquid and solid impurities, as well as gaseous products of thermal decomposition and radiation decomposition of the coolant, and fission products. Removal of liquid contaminants is based on the mass transfer principle, while gaseous impurities are removed on the basis of deaeration and freezing-out principles. Corrosion products and other solid particulate matter are picked up in mechanical filters. The coolant is bled from the condensers for purification in amounts of approximately 1% of the total flowrate.

In addition to the above-mentioned systems for normal operation of the reactor, there are the evacuation system 18, make-up system 20, water-supply system, and several others.

A fired boiler 29 was connected to the stream, parallel to the reactor, in order to shorten the time required to debug the process equipment of the primary loop and auxiliary systems to facilitate operation of the systems in the prestartup period, and also to get one of the subloops working with nonradioactive coolant on stream. This 15 MW (th) boiler made it possible to operate one of the subloops at normal coolant parameters and with only a partial load on the turbine.

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LIST OF REPORTS PRESENTED TO FIRST PANEL OF THE COMECON '70 CONFERENCE

Physical Characteristics of the V-3M Reactor – G. Ya. Andrianov et al. (USSR)

Experiments designed to determine the physical characteristics of uranium–water lattices, and carried out on a test stand at the I. V. Kurchatov IAE [Institute of Atomic Energy] and at the Novaya Voronezh' nuclear power station [NVAES] when the second power unit was started up, are described.

Results of measurements of the dependence of the geometric shape factor B_0^2 and of the effectiveness (worth) of control rods on the concentration of boric acid in the moderator, at t = 20°C, are presented. Control rod worth values were measured for three variants of a fully scaled-up core. Results of core safety experiments are also cited.

Experiments designed to determine control rod worth (both differential and integrated worth values) as a function of moderator temperature and as a function of the concentration of boric acid in the moderator, carried out directly during the startup of the second power unit at the Novaya Voronezh' nuclear power station, are also described.

The experimental data obtained are of utmost value for correcting computational procedures in the design of reactors of this and similar types.

New Results on Research Carried out by the Rossendorf Central Nuclear Research Institute in the Field of Reactor Physics (GDR [German Democratic Republic (East Germany)])

Research work done at the Central Nuclear Research Institute in the field of reactor physics has been geared to the solution of problems in procedures and techniques. Particularly noteworthy are: 1) theoretical work: in expanding the potential applications of the Monte Carlo method applied to nuclear reactor design calculations; development of computer programs for computing correlation functions; calculations of the Doppler effect based on transmission functions; 2) experimental work: design of a pile oscillator with reactivity sensitivity $\Delta\rho \leq 10^{-8}$; testing out statistical techniques in the measurement of negative reactivities; looking into possibilities of improving means of monitoring the on-stream performance of power reactors in nuclear power stations through the use of dynamic methods; investigations of the dependence of kinetic effects in the annular core of the type RRR reactor at Rossendorf on the specific site in the core.

Basic Operating Characteristics of the Reactor of the Second Power Unit of the Novaya Voronezh' Nuclear Power Station – L. M. Voronin, F. Ya. Ovchinnikov, S. N. Samoilov, Yu. V. Malkov, V. K. Sedov, Yu. V. Markov, A. S. Dukhovenskii, and A. I. Belyaev (USSR)

Translated from Atomnaya Énergiya, Vol. 30, No. 2, pp. 185-191, February, 1971.

Operating Experience of the Rheinsberg Nuclear Power Station – Z. Mendel (GDR)

Investigation of the Radiation Environment at the First and Second Power Units of the Novaya Voronezh' Nuclear Power Station – V. N. Mironov, G. V. Matskevich, P. V. Teplov, V. A. Kazakov, Yu. A. Akkuratnov, S. N. Samoilov, Yu. K. Gladkov, and V. B. Dobrynin (USSR)

Results of experimental investigations are cited. A theoretical analysis is presented of processes governing the formation of the radiation environment at the nuclear power station under different sets of operating conditions.

The report takes up the following topics: biological shielding and devising a special container to facilitate preventive inspection of the reactor pressure vessel of the first power unit of the Novaya Voronezh' nuclear power station; investigation of processes by which fission products from nuclear fuel in the primary loop of the VVER reactor build up, and development of a mathematical model of these processes; effect of coolant cleanup on the formation of radioactive deposits throughout the primary loop, and analytical dependences governing the pattern of buildup of radioactive Co^{60} deposits in the primary-loop system.

A set of measures for improving the radiation environment under normal operating conditions of the nuclear power station, and also in preventive inspection of equipment and in repair and maintenance work, is recommended on the basis of the research done and on the basis of calculations.

Development Outlook for Pressure-Vessel Boiling-Water Reactors with Natural Circulation of Coolant – I. N. Sokolov, V. I. Barybkin, I. A. Pilat'ev, and V. P. Evstigneev (USSR)

The hydrodynamics of the natural circulation circuit for a pressure-vessel type reactor are discussed. It is shown that recourse to artificial measures to smooth out the power release pattern makes it possible to attain average specific burnup levels as high as 60 to 70 kW/liter. It is also demonstrated that the limiting factor at high average steam content levels is the margin to burnout, rather than resonance instability of the core.

The recourse to internal resources of a boiling-water reactor (supplying water at different degrees of subcooling to various parts of the core, rearrangements of fuel assemblies, self-flattening of the power field according to the degree of burnup, application of artificial measures to lower the volume stream content) makes it possible to step up the percentage fuel burnup appreciably.

The VK-50 facility is used as illustrative example in correlating the actual and expected cost characteristics of a boiling-water pressure-vessel type reactor.

Effect of Various Engineering Shielding and Protection Devices at Nuclear Power Stations with Pressurized-Water Reactors on Release of Radioactive Materials to the Environment – W. Burkhardt, H. Eichhorn, and W. Schimmel (GDR)

Radioactive discharges to the surrounding territories from nuclear facilities occur under normal operating conditions, and also as a result of discharges to the surroundings in response to infrequent and extraordinary events or catastrophes, the possibility of which can never be totally excluded. The average annual quantity of discharges in response to extraordinary events must not exceed the total quantity of discharges occurring under normal operating conditions.

Possible accidents are analyzed with the aim of promoting future safety by taking the appropriate measures, with technical possibilities and costs taken into account.

Investigations of the effect of various nuclear power station components (such as the loops, reactor emergency cooldown, etc.) on radiation hazards were carried out.

Stress is laid on the need to generalize nuclear power station operating experience, in dealing with the probability of malfunctioning of various components.

Improvements in the method of accident analysis and its applications are recommended, for example for the process monitoring and measuring instrumentation, automation, electrical systems, etc.

The conclusion is drawn that the ultimate results have to include calculations of the yearly average exposure dose to the population, and this, together with the exposure brought about under normal operating conditions, must not exceed tolerable dose levels.

Distribution of Deposits and Their Activity on the Surfaces of Equipment and Piping of the Single-Loop VK-50 Reactor – L. N. Rozhdestvenskaya, Yu. G. Lavrinovich, Yu. V. Chechetkin, A. I. Zabelin, and T. S. Svyatysheva (USSR)

High-Power Steam Generators for Nuclear Power Stations with Water-Cooled Water-Moderated Reactors – V. F. Titov, V. I. Grishakov, G. A. Tarankov, V. G. Suprunov, V. P. Denisov, and V. V. Stekol'nikov (USSR)

Designs of horizontal and vertical steam generators for the power units of nuclear power stations with 1000 MW(e) VVER reactors, and four circulating subloops, are discussed. Light is shed on the basic aspects of the design of transportable steam generators of large unit power output. The groundwork is done for basic solutions acceptable in the development of the basic design components of steam generators. It is shown that a large number of experiments in the hydrodynamics of a water – steam mixture steam separation, and studies of the material strength of various component assemblies, are helpful in improving reliability while stepping up the unit power output of nuclear power station process equipment, including the steam generators. Possibilities of designing steam generators with natural circulation of boiler water and generation of superheated steam are analyzed.

Experimental Developmental Work and Operation of Several Variants of Control Mechanisms for Water-Cooled Water-Moderated Power Reactors – V. D. Shmelev, A. S. Sokolov, V. S. Anisimov, V. V. Stekol'nikov, V. P. Denisov, V. I. Naletov, and V. E. Glot (USSR)

Results of experiments on experimental development and operation of helical control mechanisms used on reactors and process equipment now operating in the USSR are reported.

The helical control mechanisms, of simple design, are capable of reliably displacing control actuators of considerable weight. For example, in the usual design of control actuators for water-cooled water-moderated reactors in use in the USSR, the actuators weigh 300 to 400 kg.

Design variants of the helical control mechanisms and ways of improving these design variants on the basis of operating experience with some of them, are reported.

The basic characteristics of helical control mechanisms for existing equipment, and experimental data on development and improvements in helical control mechanisms, are reported.

Measurement of Fission Fragment Concentration in the Primary Loop by the Delayed Neutron Method — J. Moravek and E. Hladky (CSSR)

A method for measuring delayed neutrons by monitoring fission fragments present in the primary loop of a gas-cooled nuclear reactor installation is described. An algorithm is derived for calculating sensitivity in detection of fission fragments. Results of calculations of the dependence of sensitivity on the basic parameters of the reactor and detector are presented. The basic parameters of the detector are optimized theoretically, and a suitable design is proposed.

An estimate is presented of the method from the vantage point of the chemical state and of the behavior of radioactive iodine in the primary loop and in the piping, sampling, and possible use of samples for back-up monitoring of leaks in the fuel elements.

KGO* Delayed-Neutron Subloop System for Fuel Elements — L. I. Golubev, V. P. Kruglov, and Yu. A. Borisov (USSR)

A description is presented of a system for delivering samples and handling secondary equipment for the KGO subloop system of fuel elements, for pressure-vessel type reactors with a rod system compensating excess reactivity.

Results of the operation of a KGO system continuously monitoring the state of fuel-element cladding in the VVER-1 reactor system, over the course of two reactor campaigns, are presented. Experiments designed to determine the core regions with leaky fuel elements by a method centered on local reductions in power output are described. The results are compared to the results for assembly type KGO arrays in shut-down equipment. Several technical characteristics of the KGO subloop system for a VVER-1 reactor are cited.

Improvements in Water Management and Heat Transfer Equipment Equipment for Single-Loop Nuclear Power Stations — T. Kh. Margulova and F. G. Prokhorov (USSR)

At the present time, the reliability of the water management system for single-loop nuclear power stations is established by 100% condensate cleanup, with the turbine condensate and discharges from all the reheaters passed through the cleanup step. Reheater tubing is made of stainless austenitic steels.

The following positive results can be attained by improving the water management system: cheaper condensate cleanup; use of brass reheater tubing; not pouring reheater discharges into the condenser, and thereby lowering the cost of the turbine installation; eliminating a cooler for the separated steam, which also lowers the cost of the installation; refraining from plating the reheater vessels; making all the condensate piping of carbon steels.

Experimental research was conducted as part of the effort to realize the proposed scheme, and attention was centered on efficient use of the anion exchange resin in an alkaline stream, and on thermal stability of the cation exchange resins.

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*KGO — FECM.

Problems Overcome in the Building of Electric Power Generating Stations in the German Democratic Republic

– K. Rambusch (GDR)

In the light of the widespread use of nuclear power stations to meet the needs of the GDR in electric power and heat power, requirements have been advanced for an international division of labor in this area, particularly in regard to work done in the GDR.

The necessary prospective requirements for the growth of the power outputs of individual power units and of the nuclear power station as a whole are presented. The effect of siting of nuclear power is also considered. Fundamentally necessary costs, for the installation and for the national economy as a whole, are pointed out.

The Concept of Radiation Safety for a Nuclear Power Station

– E. Hladky, J. Kopec, Z. Melichar, and J. Moravek (CSSR)

Exact definitions of the concept of radiation safety with application to nuclear power stations are discussed. The basic aspects of radiation safety, and methods for approaching the solution of the problem, are reviewed in terms of the type of reactor, the function of the reactor, and its siting. Fundamental problems are pointed out and their role in the overall solution of nuclear power station safety problems is evaluated. The general concept of radiation safety is put forth, special aspects of the solution of the problem and the corresponding investigative efforts for specific types of nuclear power stations are discussed.

Procedures and Periodic Monitoring of the State of Metal in Radioactive Loop Process Equipment – N. N. Shabanov and V. A. Mentsov (USSR)

Measures for achieving representative monitoring of the state of metal in loop process equipment, and for improving the operational reliability of nuclear power station equipment, have been worked out on the basis of analysis and generalization of operating experience acquired at both foreign and Soviet nuclear power stations, as well as results of investigations of nuclear power stations already in service or recently brought into service.

Information is provided on investigations of the inner surfaces of two top shells for the pressure vessel in the water-cooled water-moderated reactor of the first power unit in the Novaya Voronezh' nuclear power station, including the sealing surface of the principal joint, and the inner surface of the "hot" reactor piping connections.

The investigative procedure is based on the use of an inspection booth, which offers biological shielding to protect the personnel working in the booth from radioactive objects, and also makes it possible to prepare the surface of the pressure vessel for examination and analysis by nondestructive testing methods. The following methods were used in the work: visual inspection (some parts of the equipment were photographed), dye-penetrant methods, and eddy-current methods.

Results of an examination of the second and third power generating units in the Novaya Voronezh' nuclear power station are cited.

After a "hot run-in" of the primary-loop process equipment in the first power unit, visual inspection was performed and the state of the metal was investigated by various methods.

The procedure used and the results of an examination of the pressure-vessel in the VZ-M equipment were reviewed. An ultrasonic pulse-echo method was used to inspect welded joints in the shells of the vessel and in the bottom closure of the equipment, as well as the ring seal on the vessel cover. An ultrasonic reflection-shadow method was used to examine welded joints on pipe connections. Information on the results of an examination of welded joints in the basic circulation piping, made of austenitic steel, by a visual inspection method and the dye-penetrant method, was also reported. Research on development of procedures for inspecting bends in the basic circulation piping, made of austenitic steels, has been carried forward.

The preliminary procedure in the investigations, including visual inspection, ultrasonic nondestructive testing, and dye-penetrant nondestructive testing, of the metal, are cited in detail in the article.

Some Safety Problems for Nuclear Power Stations with VVER Reactors – V. F. Ostashenko and I. I. Bumblis (USSR)

Hypothetical Accident at a Nuclear Power Reactor with a Pressure-Vessel Type Reactor – B. A. Dement'ev and V. D. Kuznetsov (USSR)

Results of an investigation of the consequences of an accident presumably due to a burst of the circulation loop at a nuclear power station with a pressure-vessel type reactor are reported. Analytical models making it possible to secure the needed information for subsequent analysis of the post-accident radiation environment are discussed (pressure fluctuations in the primary loop and in the reactor rooms; temperature conditions of the fuel elements; time behavior of core meltdown).

Some aspects of the progress of an accident of this type are discussed. Some of the results of an experimental investigation of this type of accident processes, carried out on a model, are reported. The acceptability of the approach used for analysis of the consequences of accidents of this type is confirmed. The results obtained can be used in evaluating the consequences of a presumed accident occurring in the primary loop of the nuclear power station with a VVER reactor.

Calculations of Changes in the Parameters of the Primary Loop of a Nuclear Power Station with Water-Cooled Water-Moderated Reactors in the Event of Piping Bursts – B. K. Mal'tsev, D. Kh. Khlestkin, L. V. Stavritskaya, and V. P. Kanishchev (USSR)

An analysis is given of emergency conditions brought about by leaks or ruptures occurring in components of the primary loop in a nuclear power station with water-cooled water-moderated reactors. Data generalizing experimental research on determinations of critical flowrates for hot water are cited. Equations for conservation of energy and mass, and the equation of state, making it possible to determine the pattern of variation in coolant parameters as leaks appear in the loop, are presented on the basis of a multiregion model of the primary loop.

Calculations of variations in pressure, flowrate, and weight level in the equipment, in response to rupture of the Du-500 main circulating piping in an installation with a pressure-vessel type reactor, are cited as illustrative examples. A comparison is offered with similar data reported in the USA.

Prospective Applications of Complexing Chelates in Nuclear Power – T. Kh. Margulova, N. G. Rassokhin, S. A. Tevlin, Yu. E. Levedev, and S. V. Bogatyreva (USSR)

Several topics are taken up in this paper: 1) application of sequestering agents and compositions based on them for decontamination operations before startup and during regular service, the advantages offered by such agents, and results of full-scale on-stream applications; 2) improving the corrosion resistance of structural materials by treatment with chelating agents (in the case of pearlitic steels, cladding materials), results of research on subloops, in autoclaves in industry (operating and test-stand modes); 3) effect of chelating treatment of hydrogenation of steels; 4) use of chelating agents to correct water conditions, results of research on subloops and in capsules; 5) further development of research work in the field of applications of chelating agents in nuclear power.

Some Topics Concerning the Physics of Reactors in the Belyi Yar Nuclear Power Station – Yu. I. Mityaev, V. K. Vikulov, and V. M. Shuvalov (USSR)

Methods for Startup and Operation of Channel Type Uranium – Graphite Reactors with Tubular Fuel Elements and Nuclear Steam Superheat – P. I. Aleshchenkov, G. A. Zvereva, G. A. Kireev, G. D. Knyazeva, V. I. Kononov, L. I. Lunina, Yu. I. Mityaev, V. P. Nevskii, and V. K. Polyakov (USSR)

Design Features of a High-Power Uranium – Graphite Channel Type Reactor – N. A. Dollezhal', Yu. M. Bulkin, K. K. Polushkin, Yu. N. Klement'ev, V. V. Rylin, V. I. Krylova, G. M. Kondrat'ev, I. I. Grozdov, and K. V. Petrochuk (USSR)

A heterogeneous boiling-water type channel reactor, burning uranium fuel and graphite-moderated, designed for generation of electric power, is described. The electrical power output rating is one million kW. Water and a water – steam mixture are used as coolant. The reactor installation includes the reactor, a multiple forced circulation loop, a loop for the reactor control and protection system, the gas loop, a side shield cooling loop, a drainage system, a system cooling the holding tank water, and other ancillary systems.

Materials approved for long-term operation of nuclear reactors are employed as basic structural materials for the reactor assemblies and subassemblies.

The biological shielding of the reactor meets the public health regulations for the radiation environment during normal operation, in all the rooms accessible to personnel around the reactor pit and in the central hall.

The heat transfer monitoring system puts out recorded data on the parameters characterizing reactor operating conditions.

Radiation Safety Aspects in the Design and Operation of Channel Type Power Reactors – A. P. Veselkin, V. P. Sklyarov, Yu. E. Khandamirov, and A. I. Yashnikov (USSR)

Design of Uranium – Graphite Channel Type Reactors with Tubular Fuel Elements and Nuclear Steam Superheat – N. A. Dollezhal', P. I. Aleshchenkov, Yu. V. Bulankov, and G. D. Knyazeva (USSR)

Note on Ways of Increasing the Unit Power Output of Nuclear Power Station Power Generating Units – N. A. Dollezhal', B. B. Baturov, Yu. I. Koryakin, V. A. Chernyaev, and I. I. Zakharov (USSR)

It is demonstrated that the discontinuity found in the nuclear power industry between the unit power outputs of a reactor and steam-driven turbine sets operating in unison with the reactor have been increasing. The negative economic effects stemming from this disproportion will become particularly strongly felt in the near future, when the problem of labor costs in operating nuclear power stations will loom large, along with capital investment costs.

Possible ways of raising the unit power output of steam turbine installations are analyzed. The use of low-temperature binary cycles leading to the highest increase in unit power (ten times, theoretically) is demonstrated.

Results of cost savings comparison estimates for nuclear power stations with uranium – graphite thermal reactors using steam – water turbine sets and water – freon turbine sets are presented.

Binary turbine sets of large power output will make it possible to improve the engineering cost picture of nuclear power stations appreciably.

Management of Water and Chemicals at a Nuclear Power Station with a Channel Type Reactor and Nuclear Steam Superheat – O. T. Konovalova, T. I. Kosheleva, V. V. Gerasimov, L. S. Zhuravlev, and G. A. Shapov (USSR)

Some Topics Concerning Water Management at Nuclear Power Stations with Boiling-Water Reactors, and Selection of Structural Materials – B. A. Alekseev, V. A. Ermakov, and V. F. Kozlova (USSR)

The basic principles guiding the selection of water treatment conditions for single-loop nuclear power stations with boiling-water reactors are discussed. A neutral, i.e., an "uncorrected," water management setup characterized by the value $pH_{25°}$ = 6.5 to 7.0, and by avoidance of any means for suppressing oxygen formation, is recommended on the basis of experiments carried out on subloops of the MR reactor at the I. V. Kurchatov Institute of Atomic Energy [IAE]. This approach is the basic one for new nuclear power stations with boiling-water reactors now being planned in the USSR. It was established, through measurements taken on the subloop of the MR reactor, that the rate of efflux of gases originating in radiolysis of water in the transition from pressure-vessel type reactors to channel type reactors (the type used in the Belyi Yar Nuclear Power Station) decreases by several factors.

Analysis of literature reference data and results of test stand and loop studies, carried out in the Soviet Union, demonstrate the applicability of pearlitic steels, in principle, to fabrication of some components in the reactor loop, and in the steam flow lines and passages for condensate and feedwater.

Arguments are advanced on proper organization of cleanup of blowdown water and feedwater, relying on ion exchange beds and mechanical filters, either the washed-on type or electromagnetic type.

Note on Performance of Ion Exchange Filters at Single-Loop Nuclear Power Stations – Yu. V. Chechetkin, A. I. Zabelin, L. N. Rozhdestvenskaya, V. D. Kizin, E. K. Yakshin, and L. N. Masnaya (USSR)

A design arrangement is proposed, and estimates of integral exposure dose are cited, for determining the prospective service life of ion exchange resins in service in turbine condensate cleanup systems, for different cases of fuel-element rupture in a single-loop nuclear power station based around a boiling-water reactor. Radiation fields in the vicinity of ion exchange filters are calculated. Agreement between predicted data and experimental data is noted.

One year's operating experience with a condensate cleanup facility at a nuclear power station using a VK-50 reactor is discussed. Changes in hydrodynamic characteristics with time on stream are reported. Effectiveness in removing contaminants from water on cation and anion exchange resin filters, under normal station operating conditions, and under emergency conditions in the turbine condenser, is estimated. Some recommendations are put forth regarding the use of condensate cleanup systems of the type.

Program for Estimating Nuclear Power Development in the CSSR and Some Computational Results – S. Novak and T. Rajci (CSSR)

The complexity of nuclear power station development problems calls for detailed treatment of some of the factors governing the effectiveness of any path chosen toward the goal. A computational program has been developed in Czechoslovakia for estimating the effects of distinct factors on the development of nuclear power stations in the CSSR. The basic characteristics of the program, the input data, and the progress of computations, are covered. The effect of the choice of initial power station developmental period is estimated for full scale-up and cost calculations. Attention is given to combined power generation by nuclear power plants in conjunction with conventional electric power generating plants (changes in utilization time, dynamic properties required, etc.). Results of calculations for different variants of the initial developmental period for full scale-up (covering 50 years) and cost calculations (covering 20 years) are cited.

Preparation for Startup of the A-1 Nuclear Power Station – E. Horvat (CSSR)

Methods for Detecting Burst Fuel Elements in the Gas-Cooled Reactors, and Theoretical System Optimization – E. Hladky, Z. Melichar, and J. Kubik (CSSR)

A critical analysis is carried out, on the basis of operating experience with foreign gas-cooled reactors, of various methods, and some possible solutions for the design of a system capable of detecting burst fuel elements are proposed. Attention is focused on additional monitoring of coolant, and methods for processing the detection system data.

General criteria are formulated for the design of optimum systems, and possibilities of theoretical optimization of systems are demonstrated, accompanied by concrete computational findings. The relationship between the time behavior of the failure detector signal and the time of fuel-element failure or damage is discussed in the light of calculations of detector sensitivity.

Fuel Cycles for Water-Cooled Water-Moderated Thermal Reactors – S. M. Feinberg and I. K. Levina (USSR)

The distinctive feature of the work discussed in this article is the treatment of a long sequence of different campaigns involving deep fuel burnup in pressure-vessel reactors of the pressurized-water type, for specific VVER reactor operating conditions. As many as 30 burnup campaigns are covered, thereby providing a fairly detailed concept of the isotope composition and natural uranium needs for assessing the long-term developmental outlook of large-scale nuclear power.

The calculations were carried out for an "ideal" water-cooled water-moderated reactor, and also for an "ideal" radiochemistry and metallurgy cycle. This means, first of all, that a pressure-vessel reactor with a constant nuclear refueling system is proposed, and secondly that the radiochemistry and metallurgy cycle features zero fuel time lag and zero fuel losses.

This treatment of the problem determines the physical minimum of natural uranium needs. Corrections to be introduced into the "ideal" fuel cycle are clarified.

Fuel cycles using uranium dioxide, metallic uranium, and thorium compositions with U^{235} and U^{233} are discussed. These compositions make it possible to evaluate the outlook for use of thorium in order to save on natural uranium.

An alternative considered is bringing fast breeders into play in order to economize on natural uranium. This makes it possible to estimate the doubling time of breeder systems and of the radiochemical metallurgical reduction process through which these savings can be realized.

Investigations in the Field of Nuclear Power Station Fuel Cycles, Conducted in the Chemical Sector of the Nuclear Research Institute of the Czechoslovak Academy of Sciences – S. Havelka, L. Berak, V. Kourim, J. Peka, M. Podest, and V. Srajer (CSSR)

Investigations were concerned on three salient problems of the course of the past five years.

1. Methods for the production of sinterable uranium dioxide, and powder compactification techniques, were studied. Preliminary investigations were conducted on uranium monocarbide and uranium carbonitrides, yielding some interesting findings on the chemical behavior of these materials. Investigations of the physicochemical fundamentals of the "sol – gel" method were initiated.

Further work was directed toward setting up an experimental base for effecting a transition to investigations of fast-reactor fuel.

2. A modified purex method for reprocessing spent fuel from VVRS type reactors was worked out on a laboratory scale. Work is being completed on setting up a laboratory circuit with a throughput of 0.5 kg U per day. Planning work has begun on a scaled-up pilot facility with a capacity of 10 to 20 kg U per day.

The field of fluoride reprocessing techniques is represented by work on sorption of fluorides; a fluorination technology has been worked out in this area. Work on building a circuit with a capacity of 2 kg U per day is being completed.

3. Separation and isolation of long-lived toxic fission products, and immobilization of high-level wastes in basalt rock, are being studied.

Experience in the Construction and Startup of the BOR-60 Reactor – A. I. Leipunskii et al. (USSR)

Research Findings on the Operating Characteristics of the BOR-60 Reactor during Power Startup – O. D. Kazachkovskii, V. A. Afanas'ev, E. V. Borisyuk, V. M. Gryazev, V. N. Efimov, V. P. Kevrolev, V. I. Kondrat'ev, N. V. Krasnoyarov, S. A. Markin, and A. M. Smirnov (USSR)

Results of experiments on the preparation of the BOR-60 reactor installation for the power startup are reported. The stable operating region of the pumps in parallel streams, with the performance of check valves taken into consideration, is determined. Data are obtained on specific heat and power losses in the primary loop, so that the hazard of various sets of emergency conditions that might occur when the BOR-60 reactor installation is on full power can be estimated.

During the ride up to full power, and during the first stage of normal operation, experiments were conducted to study natural circulation, emergency cooldown performance, the characteristics of the heat exchange equipment, and so forth. Transients accompanying energizing of emergency protection systems were optimized. Transients occurring in response to disturbances in the basic system variables were obtained. The self-regulating properties of the reactor installation are illustrated.

The ZRR Fast Experimental Reactor, and Proposals for its Experimental Utilization – M. Pasek, Z. Tluchor, V. Chlumsky, J. Cermak, F. Dubsek, J. Hrdlicka, A. V. Karpov, M. F. Troyanov, M. I. Kulakovskii, V. I. Matveev, et al. (CSSR and USSR)

The ZRR experimental fast reactor, with power ratings to 60 MW, went into planning stages in Czechoslovakia in 1969 (with consultation with staff members of the Obninsk Power Physics Institute [FEI]).

The parameters of the ZRR reactor are close to those of the BOR reactor.

But the ZRR reactor differs from its BOR counterpart in several ways: there are two sodium subloops with 110 mm diameter field type channels located in the reflector; the proposed neutron flux is $1.2 \cdot 10^{15}$ neutrons/$cm^2 \cdot$sec. The channel can accommodate an experimental fuel assembly with 19 fuel pins. The closed-loop subloop design arrangement makes it possible to achieve operating conditions in the experimental channel (sodium operating temperature, contamination of coolant by fission products, effect of thermal shocks, limiting states of fuel elements, leak testing of fuel elements). There are two channels at the center of the core, for experiments on specimens of structural materials and fuel materials at peak neutron flux. The subloops and channels must be equipped with modern sophisticated instruments in order to carry out the indicated experiments.

The design of the ZRR reactor and its fuel handling system are described. Physical characteristics and the general process flowsheet of the ZRR reactor installation are given.

There are several reports devoted to the possibilities for conducting experiments on the ZRR reactor, and on further work along the lines of the fast reactor research program, with the ZRR reactor serving as research tool.

Measurement of Sodium Flowrate and Sodium Fill Level in Fast Power Reactors – V. D. Taranin et al. (USSR)

Exact measurement of sodium flowrate by means of large electromagnetic flowmeters, and measurements of sodium fill level at high temperatures, over the range from 0 to 6000 mm, are discussed. The layout of a metrological flowmetering stand which can be used to calibrate and certify magnetic flowmeters with their upper measurement range extending to 4000 m^3/h liquid sodium is presented, and the basic requirements and specifications for the components of this stand are formulated. The design of a high-temperature sodium level gage, and its arrangement and siting, are described. Results of adjustment operations and service-life tests of the instrument on the sodium test loop of the BR-5 reactor are reported.

Final Work on Pump Designs for Power Plants Using the BN-350 and BOR-60 Fast Reactors – F. M. Mitenkov, E. N. Chernomordik, V. I. Sharonov, M. I. Mikhailov, V. M. Budov, and É. G. Novinskii (USSR)

Results of adjustments done on liquid-sodium pumps for service in nuclear power plants using the BN-350 and BOR-60 fast reactors are reviewed.

Design layouts for circulation pumps for the primary and secondary loops of the power plants referred to, and the basic performance characteristics of the pumps, as well as the materials used, are duly reported. Light is shed on experimental work done on the flow passages of the pumps, on bearing assemblies, and seals for the rotating pump shaft.

Procedures, conditions, and results of pump-engine sets tested on water streams and liquid sodium streams are presented. Topics relating to the performance of pumps in starting and operating transients, determination of axial forces under different sets of operating conditons, and entrainment of gas on the delivery lines with leakage and carryover of basic into the primary loop, are discussed, with analysis and experimental verification of possible ways that carbon might get into the loop from the pumps.

Fabrication errors occurring when pumps are being finish-machined, and measures taken to eliminate them, are described.

Effectiveness of Fuel Utilization in Fast Power Reactors – V. B. Lytkin, A. I. Leipunskii, V. V. Orlov, and M. F. Troyanov (USSR)

The limitations on comparatively cheap uranium resources may bring about a rise in the cost of electric power generated by nuclear power stations, if nuclear power developmental policies and reactor characteristics are not selected properly in the development of the nuclear power industry. Fast power reactors therefore have a crucial role to play in economies of natural uranium resources. It is demonstrated how the resulting rates of consumption of fissionable materials at different industry growth rates are affected by such reactor characteristics as specific loading, percentage burnup, breeding ratio, and external fuel cycle time.

The natural uranium rates are also effected by the type of uranium reactors used for initial buildup of plutonium supply and for subsequent combined operations (where required) in conjunction with fast plutonium-fueled reactors are preferable to thermal ordinary-water reactors in this regard. Data on consumption of natural uranium in the development of the nuclear power industry in the USSR, extrapolated to the year 2000, are presented up to the 600 million kW level.

Investigations of Physical Characteristics of the BOR-60 Reactor – O. D. Kazachkovskii, I. N. Alekseev, S. M.

Baranov, G. I. Gadzhiev, V. M. Gryazev, Yu. M. Karatsuba, N. V. Krasnoyarov, V. N. Lychagin, and V. S. Fofanov (USSR)

Results are reported on theoretical and experimental investigations of the basic physical characteristics of the BOR-60 fast reactor.

Critical core parameters, worth of control rod bundles, variations in reactivity in response to refueling with new fuel assemblies, spatial distribution of power density, and the neutron flux for different energies, were measured during the processes of physical startup and power startup of the reactor. The temperature and power effects of reactivity, and the fuel burnup effect, were also investigated.

It is demonstrated that changes in coolant flowrate and in coolant pressure in the reactor gas chamber drastically affect reactivity, and are governed by changes in the size of gas bubbles present in the sodium stream and in the working fuel assemblies.

Results of measurements of neutron flux and gamma-ray flux in experimental fuel assemblies and in the biological shielding are also reported.

Experimental Investigation of Activity Distribution in the BOR-60 Reactor – V. M. Gryazev, N. V. Krasnoyarov, Yu. V. Chechetkin, G. I. Gadzhiev, V. I. Polyakov, V. S. Fofanov, I. G. Kobzar', Yu. I. Leshchenko, V. V. Konyashov, and V. N. Rybakov (USSR)

Experimental investigation of the radiation distribution in reactor shielding is discussed. Activity levels in the sodium coolant, and radiation fields in the production rooms of the reactor installation, are estimated. Results of investigations of the radio isotope composition in the gas chambers of the loop are reported.

The results obtained can be used to verify shielding design procedures and procedures for calculating activity distribution over the technological loops of fast reactors, and also in the solution of other practical problems.

Note on Assignment of Parameters for Nuclear Power Stations with High-Output Fast Reactors – O. D. Kazachkovskii, N. V. Krasnoyarov, R. V. Nikol'skii, E. A. Grachev, T. M. Ziganshin, E. V. Kirillov, and R. E. Sekletsova (USSR)

Computational Methods for Estimating the Stage of Damage to a Sodium Loop in Severe Failure of Tubing in a Sodium – Water Steam Generator – V. M. Poplavskii, Yu. E. Bagdasarov, and V. H. Leonchuk (USSR)

Computational methods developed by the authors for estimating the effects of sodium – water interaction directly in the reaction zone, and also on various portions of the sodium loop, in the event of severe failure of heat-transmitting tubes in a steam generator, are presented.

A description is given of a physical model of the process for use in constructing design computational procedures. Maximum possible emergency pressures and temperatures that can occur in a steam generator immediately in the reaction zone are analyzed. A procedure for calculating reaction parameters in their time variation, in the neighborhood of tube failures, with coolant compressibility taken into account, is described.

Methods for calculating hydrodynamical phenomena in the sodium loop of the installation, and in the emergency handling systems of the steam generator unit, are presented. Simplified methods are outlined for calculating emergency parameters convenient for use in the stage of tentative determination of the basic characteristics of the liquid-metal system.

Calculated values obtained by different procedures are analyzed and compared with experiment.

Experimental Investigations of Methods for Detecting Fuel Assemblies with Failed Fuel Elements in the BR-5 Reactor System – N. N. Aristarkhov, I. A. Efimov, L. I. Mamaev, M. P. Nikulin, and V. S. Filonov (USSR)

A system capable of checking leaktightness of fuel assemblies extracted from the core of the BR-5 fast reactor and transferred to storage is described in this article. Basic results of two mass checks run on the leaktightness of working fuel assemblies with PuO_2 fuel elements, carried out in 1961 and 1965, are reported. Some of the regularities governing escape of gaseous fission products from failed fuel elements are established.

Results of tests on the integrity of fuel assemblies with PuO_2 fuel elements, extracted from the reactor and placed in cans with molten lead, are reported. Unloading and storage techniques are described, with methods for monitoring leaks in fuel bundles containing UC and PuO_2 fuel elements in cans with molten sodium, and the results of checks made outside the reactor on bundles under a layer of sodium, in the monocarbide zone of the BR-5 reactor, are also reported.

Some Aspects of Fast-Reactor Fuel Cycle Optimization – F. M. Mitenkov, G. B. Usynin, V. A. Chirkov, V. A. Shibaev, and E. A. Shlokin (USSR)

Results of investigations carried out on selection of dimensions for the fuel bundle, and fuel reloading conditions for a fast power reactor are presented. The effect of repeated refuelings of the core and the length of the interval of continuous reactor operation on the amount of fuel involved in the cycle, the breeding ratio, and the doubling time, is analyzed.

It is shown that an orientation to very infrequent refuelings runs the risk of deteriorating fuel cycle economies, and this is crucial in the case of reactors with an internal conversion ratio that is not high to begin with. Comparisons are made between overheating of the fuel-element cladding due to increased fuel bundle dimensions, and overheating due to other factors (specifically, off-optimum positioning of reactivity controllers in the core, and replacement of absorbing compensator rods by fuel rods). Ways of improving reactor performance by rational utilization of fuel discharged from the reactor are discussed.

The BRG-30 Pilot Scale Power Generating Installation, with a Gas-Cooled Reactor Using Dissociating Coolant – A. K. Krasin, V. B. Nesterenko, L. I. Kolykhan, B. E. Tverkovkin, V. P. Slizov, Yu. V. Shufrov, V. P. Bubnov, B. I. Lomashev, and M. E. Gorodetskii (USSR)

Sodium Steam Generator Rated at 500 MW(th) – V. V. Stekol'nikov, Yu. E. Bagdasarov, V. F. Titov, P. N. Bogdanovich, L. F. Fedorov, E. V. Kulikov, and V. M. Poplavskii (USSR)

The design and specifications of a through-flow type unfired-vessel steam generator with a thermal output rating of 500 MW, consisting of two evaporator and steam reheater vessels, are discussed. The flowsheet and startup conditions of the steam generator are shown, its basic operating conditions are analyzed, and attention is given to requirements imposed on the automatic control system, measuring and monitoring instrumentation, and interlocks for safe operation of the steam generator and of the power station as a whole.

The basic stands taken on the nomenclature of research and development work done as groundwork for this design of the steam generator are reviewed in terms of heat transfer, hydraulics, and materials strength.

DEVELOPMENT, INVESTIGATION, AND RADIATION STABILITY OF STRUCTURAL AND FUEL MATERIALS FOR NUCLEAR POWER STATIONS

PROCEDURES AND TECHNIQUES FOR TESTING FUEL ELEMENTS IN SUBLOOP CHANNELS OF THE SM-2 AND MIR REACTORS

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The SM-2 and MIR research reactors [1-3] are designed for testing fuel compositions, prototypes of fuel elements, structural materials and scaled-up fuel assemblies, for power reactors or research reactors in the design stage. The SM-2 and MIR reactors are provided with various subloops to expedite this type of work, and each of these subloops boasts two or more experimental channels.

The basic characteristics of the subloops are listed in Table 1, where we introduce the following notation: LTWSL for the low-temperature water subloop; HTWSL for the high-temperature water subloop; BWSL for the boiling-water subloop; HTW for another variant of high-temperature water subloop. The positioning of the channels in these subloops, within the reactors, is indicated in Figs. 1 and 2.

The high neutron flux and the relatively small experimental volume of the channels in the SM-2 reactor are utilized most conveniently for studying fuel compositions and for studying prototypes of future fuel elements. The MIR reactor is best suited to service-life testing of fuel assemblies under conditions approximating fairly closely the actual operating conditions of the reactor being designed. Despite these features, problems of procedure relating to testing in these reactors have a lot in common.

Problems of Procedure in Subloop Testing

The organization and execution of subloop experiments conform to the following sequences of stages: formulation of the problem and tentative choice of exposure site (type of reactor, type of subloop and channel); selection of an existing design or development of a new design of the prototype for the proposed irradiation device; fabrication of the prototype; taking neutron physics measurements with the prototype against a physical model of the corresponding reactor; more precise siting of the exposure and selection of a method

TABLE 1. Characteristics of the Loop Reactors SM-2 and MIR

Characteristic	SM-2 reactor		MIR reactor		
	LTWSL	HTWSL	BWSL	HTW	sodium
Subloop power rating, kW	3000	250	2000	2000	2000
Number of channels	7	2	2	2	2
Power output of single channel, kW	150—1000	120	1000	1000	1000
Channel experimental volume, liters	0.9—2.0	0,35	3,1	3,1	1,25
Unperturbed fast flux, neutrons/cm^2·sec	$(0,2-20,0)\cdot 10^{13}$	$5\cdot 10^{12}$	—	—	—
Unperturbed thermal flux, neutrons/cm^2·sec	$(0,8-33)\cdot 10^{14}$	$(0,5-1,1)\cdot 10^{14}$	—	—	—
Peak power production per gram U^{235}, kW/g	—	—	6,65 *	6,65 *	6,65 *
	—	—	3,40†	3,40†	3,40†
Coolant pressure, atm	50	200	200	200	15
Coolant temperatures, °C					
at entrance to channel	50	300	300	300	200—300
at exit from channel	100	330	350	350	550

* For channels in second row.
†For channels in fourth row.

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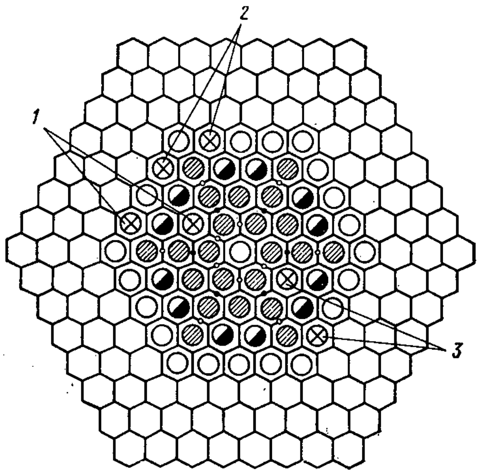

Fig. 1

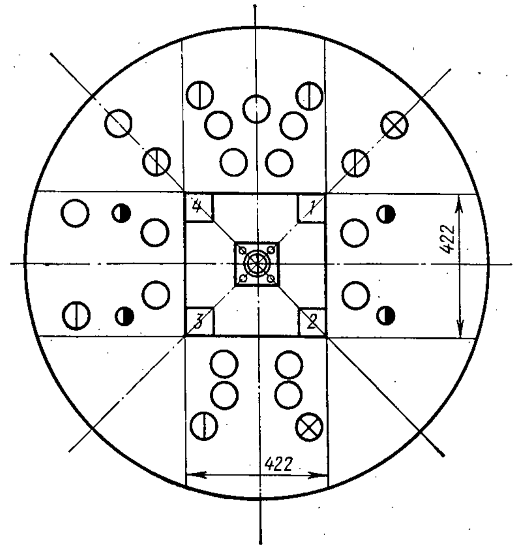

Fig. 2

Fig. 1. Schematic diagram of core loading pattern in MIR reactor: ⬡) reflector cells; ⊘) reactor working channels; ◕) rods with extra load; ◦) compensating (shim) rods; •) scram rods; ⊗) subloop channels: 1) HTW channels; 2) BWSL channels; 3) sodium subloop channels.

Fig. 2. Positioning of channels in the SM-2 reactor core: ⦶) LTWSL channels; ⊗) HTWSL channels; ○) remaining channels; ◑) control rods.

for producing the required exposure conditions; carrying out reactor subloop tests; obtaining more precise information on exposure conditions and verifying conformity between these conditions and the problems set up; design and fabrication of the irradiation device and tests prior to installation in the reactor core; materials research on irradiated specimens and compilation of an engineering report on subloop testing results. Each of these stages must be carried out while observing requirements applicable to procedures, the principal ones of which will be detailed below.

Tentative Assessment of Exposure Site and Formulation of the Experimental Problem

The fuel elements used in reactors of different types are dissimilar, as a rule, in their design, in the composition of the materials incorporated in them, in their fuel content, in operating conditions, and in various other characteristics. Moreover, subloop channels exert an important and mutual effect on each other within the confines of the small cores of research reactors. By the time a concrete experiment is set up and executed, the reactor channels are already loaded with test specimens, and the operating conditions of the reactor correspond to the operating conditions of existing channels.

The formulation of the problem in the experiment, and the choice of exposure site in the reactor, should not (at least in principle) interfere with these matched reactor conditions, as we see. (Of course, given the importance or the need to execute some specific experiment, these conditions can be eliminated either completely or for a specified time interval.) Both the representativeness of the experiment as a whole (volume of information obtained through the experiment, accuracy of that information, and so on), and the costs incurred in carrying out the experiment, plus the time taken up in work preparing the experiment, depend in large measure on the correctness of the selection of reactor type and on the selection of the corresponding subloop and experimental channel.

The experimental conditions (heat loading, temperature conditions of irradiation, burnup, coolant quality) must be brought as close as possible into consonance with the actual operating conditions of the fuel element in the core of the reactor for which it is intended. If it is not possible to bring about certain operating conditions of the fuel assembly in the projected future reactor for any reason, then the effect of such deviations from actual operating conditions on the functioning of the fuel element, or the fuel assembly

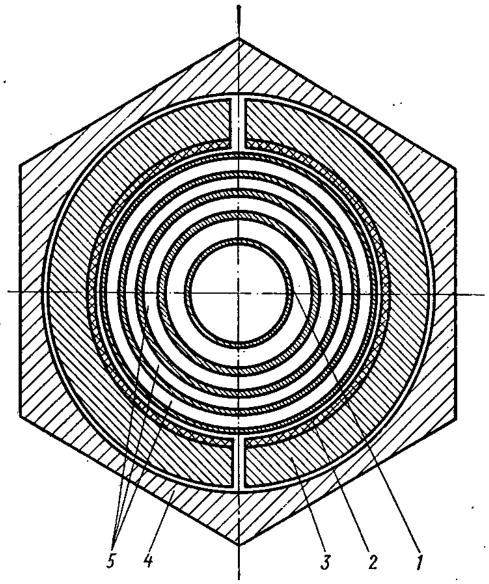

Fig. 3

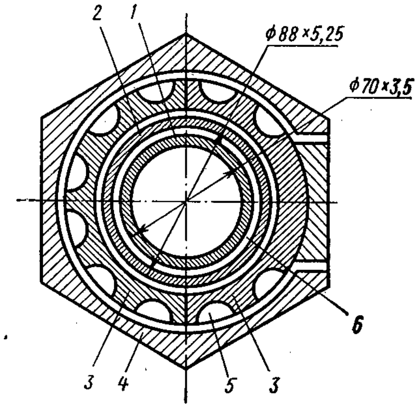

Fig. 4

Fig. 3. Subloop channel with cadmium shield: 1) channel body; 2) cadmium shield; 3) removable half-slugs; 4) insert slug; 5) gas-vacuum chambers.

Fig. 4. Subloop moderator unit: 1) channel body; 2) channel jacket; 3) beryllium half-slugs made of beryllium; 4) guide slug; 5) absorber loading channels; 6) vacuum clearance.

as a whole, must be assessed by numerical means or by staging an additional experiment to fill the gap. On the other hand, the conditions required for carrying out the experimental problems formulated must be reproduced in the type of subloop for the channel design and type of reactor selected. This means that recourse will have to be had, occasionally, to redesigning and modernization of certain components of the subloop of channel, in order to bring about all the required conditions.

Selection of some existing irradiation devices is also proposed in preliminary assessments of exposure siting, if this corresponds in principle to the goal set. Otherwise, a prototype of a new irradiation device will have to be developed and fabricated after the preliminary assessment is made.

Neutron Physics Measurements and Ways of Bringing About Required Irradiation Conditions

The SM-2 and MIR reactors are provided with physical models which constitute faithful copies of the core of those reactors, with their reflector and with the control system components in place. These physical models make it possible to study the reactor operating conditions, the irradiation conditions in any channel of the reactor, changes in these conditions resulting from displacements of the control rods and accessories, etc., on physical mockups of the subloop channels, with no particular difficulties (which would certainly plague the work if the work were to be done on the completed reactor). Neutron physics measurements make it possible to ascertain the conditions which will permit most complete execution of the heat physics problems posed in the experiment, on the one hand, and will impose no exacting limitations on the reactor operating conditions, on the other hand.

Conditions of this sort are brought in the channels of the SM-2 reactor, as a rule, by varying the fuel content in the specimens tested. The interaction of the subloop channels can be neglected, and displacements of the control rods and accessories affect only some, but not all, of the channels. The small experimental volume virtually rules out any other stationary methods by which the power production levels of these channels could be affected. From this point of departure, and taking the large number of experimental channels in the SM-2 reactor, into consideration, the fuel concentration for each channel must be selected with special care.

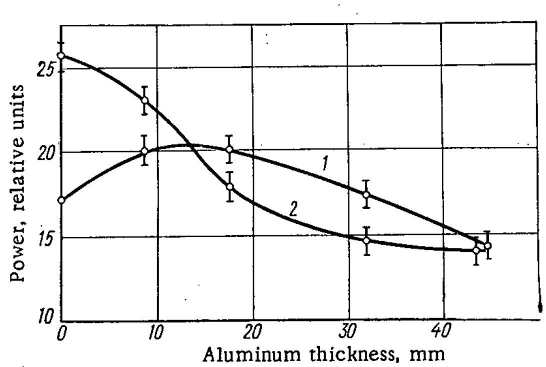

Fig. 5

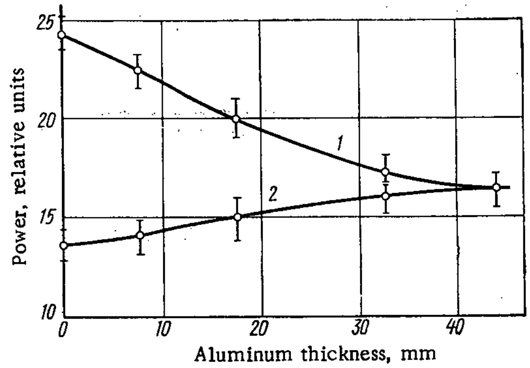

Fig. 6

Fig. 5. Variation in power output of subloop fuel assembly with thickness of aluminum in two-layer composition ("dry" channel): 1) water replaced by aluminum; 2) beryllium replaced by aluminum.

Fig. 6. Variation in power output of subloop fuel assembly as a function of aluminum thickness in two-layer composition ("wet" channel with water): 1) beryllium replaced by aluminum; 2) water replaced by aluminum.

In the case of the MIR reactor, where scaled-up fuel assemblies for future reactors are tested, the fuel concentration may not be allowed to vary over a broad range. The required operating conditions are set up in the subloop channel by individual ways of affecting power generation in the channel, when neutron physics measurements are taken. A physical model of the MIR reactor is employed to study and work out regular procedures for routine use of such control devices as the official control rods, absorbing continuous shields, and shields in the form of rodlets or wreaths of beads of different materials arranged around the subloop channel; variations in the composition of the medium in the subloop cell of the reactor lattice; changes in fuel loading patterns in the surrounding working channels.

The physical models of reactors are employed to find the power distribution over the height and cross section of the fuel assembly to be studied and tested. The results obtained are worked into heat transfer calculations, and are used in arriving at heat transfer and hydraulics conditions for subloop tests of each channel. The ability of a specific subloop to bring about the required heat transfer and hydraulics conditions over a specified power production range is looked into at the same time. If necessary, the energy production in each channel can be restricted still further, or the subloop can be redesigned.

Results of neutron physics measurements prior to the beginning of irradiation are compared as the tests progress to the readings of instruments monitoring irradiation exposure parameters. Moreover, these results are used to also estimate the redistribution of power production over the height and cross section of the test fuel assemblies during the irradiation.

In this way, measurements taken on the physical models become the first major stage in the course of the experiment's progress. This accounts for the major significance attributed to such preparatory steps as design and fabrication of the physical prototype, selection of means for controlling and adjusting the power production level, and exact studies of the neutron field over the height and cross section of the fuel assembly.

Stationary Methods for Bringing About Required Irradiation Conditions

The stationary methods [4] include those which ensure a specified power production level at the sites where subloop channels are situated for a protracted period. These methods, in addition to changes in fuel content in the fuel elements, are affected mainly by the design features of the reactor components, subloop channels, and so on.

The required irradiation conditions (heat loading, nonuniformity in power production over the radius of the fuel element in the assembly) for the channel of the sodium subloop in a cell in the second row of the MIR reactor were brought about by shielding the channel with a cadmium jacket (Fig. 3).

Vertical holes were made in the hexagonal moderator slug for some of the water channels in the MIR reactor, so as to accommodate rods of absorbing material (Fig. 4).

Control of power production by means of absorber rods makes it possible to expand the range of power levels for the fuel assemblies to be tested. In particular, in the case of a water subloop with ten rod absorbers of boron steel 8-9 mm in diameter, arranged uniformly around the subloop channel, a 1.6-fold change can be achieved in power production level. If necessary, the subloop channels in the fourth row can be shielded partially on the core side. A continuous cadmium shield has a much more drastic effect (8 to 10 times greater) on the power production level.

Note that rod type absorbers do not change fuel blockage over the fuel-assembly cross section, since the hardness of the neutron spectrum in the subloop cell undergoes no change in the process, while the cadmium shield, by increasing the hardness of the spectrum, lessens fuel blockage by roughly 1.5 times.

Another stationary method for control and adjustment of power production is changing the composition of the medium in the subloop cell. These cells are made of beryllium and aluminum in the MIR reactor. Figures 5 and 6 show how the power output of the subloop fuel assembly is affected by the composition of the medium in the subloop cell. The reported power ratios may vary depending on the design of the specific subloop fuel assembly. This method of adjusting and regulating power production leads to lower reactivity losses than when absorber rods are employed.

Operational Techniques for Bringing About the Required Testing Conditions

The necessary physical conditions for irradiation found before the experiment is carried out cannot always be sustained by stationary methods such as described. The contradictory nature of the requirements imposed on many subloop experiments, as well as the specific features of the reactors themselves, lead to a need for operational methods of affecting the power production level in the region of the core where the test specimens are positioned. The ability of the subloop to affect the temperature conditions of irradiation and the quality of the coolant is of no less significance.

Below, some of the basic methods for affecting testing parameters are described.

Operational Effect on Power. In the case of the SM-2 reactor, changes in power output in any channel involve changes in power output of the reactor as a whole, and also changes in the positions of control rods in the case of some channels. Changes in the power output of the SM-2 reactor cannot be utilized, for this reason, to control the reactor in all cases, particularly when the power runs above set point during a particular reactor campaign. Accordingly, the error for any one channel or several channels in the selection of fuel content (here we have in mind an error on the high side) will cause a lowering of the exposure parameters in the other channels and degraded performance of the SM-2 reactor as a whole.

In the case of the MIR reactor, displacements of the control rods are more autonomous in character, and as a rule cause changes in the power production level in the region where they are located. Control rods are therefore found useful for affecting the power output of reactor subloop channels. For instance, complete displacement of KS rods of the second radius may bring about changes in the power output of the HTW and BWSL subloop channels located in the fourth row of the core, by a factor of 1.3. Complete displacement of the combined KD rods (top part: absorber, bottom part: fuel assembly) changes in the power of adjacent HTW and BWSL subloop channels twofold or fivefold, depending on whether the channels are located in the second or fourth row of the core, respectively. But because of the powerful effect exerted by the KD rods on the power production level and power distribution, only KD rods sufficiently far removed from the subloop channel are employed to achieve the operational effects. Those KD rods lying close to the subloop channels are used only to bring about the required operating conditions in the initial loading, and subsequently their positions are left unaltered throughout the entire reactor campaign. It is proper to state that with the number of subloop channels currently in use in the MIR reactor, operational effects on the power output level satisfy the needs of the problems as posed, for the most part.

Operational Effects on Temperature Conditions. In a real reactor facility, the reactor power output varies over a rather broad range, while deviations from temperature conditions remain quite restricted. This highlights the importance of maintaining the required temperature conditions, when testing fuel elements, and particularly when testing fuel assemblies, as the power of the irradiated objects varies. Subloop flow arrangements must provide for maintaining specified conditions over as wide a range as possible.

The broader the range, the better the performance of the subloop, the more representative the experiments will be. The way in which temperature conditions are affected must be characterized by quick response to changes, autonomous to whatever extent possible, and sufficiently safe for the experiment itself.

The basic method for controlling temperature conditions on the LTWSL subloop of the SM-2 reactor consists in varying the flowrate of water through the channel. This method makes it possible to vary the temperature of the fuel-element cladding by ±50°C from the prevailing level at thermal loads up to (2 to 3) $\cdot 10^6$ W/ m^2. In some cases, the temperature of the water at entry to the channels can also be varied in LTWSL channels. But this method is a less effective one, because of the narrow range of variation in inlet temperature.

In the case of HTWSL subloops in the SM-2 reactor, the effect on the temperature conditions is brought about by varying the water temperature at the exit from the electric heater unit. The array of available power output steps makes it possible to effect the temperature conditions at different rate of change, in both increasing and decreasing directions. The range of these changes depends on the specified top and bottom temperature levels in testing.

In the case of sodium subloop channels, the effect on the temperature conditions of specimen irradiation is achieved in either of two ways: by varying the sodium flowrate through the channel, and by varying the sodium inlet temperature. These methods of control are easy and safe to implement, since the circulating pumps operate on direct current, allowing fine control of pump loads, and the subloop piping is provided with compartmentalized electric heater units.

Operational temperature control is achieved in the HTW and BWSL subloops of the MIR reactor by varying the water flowrate through the heat exchanger for the ion-exchange filter. This method makes it possible to maintain the temperature at the channel inlet to within ±5°C or less, if required by the experimental conditions.

Monitoring Temperature and Power Output of Fuel Assembly in Channel

In the process of reactor subloop tests, the principal parameters to be monitored are the heat loading and the temperature of the fuel-element cladding, and the temperature of the coolant upon entry to the fuel assembly. When thermocouples inserted in the fuel assemblies are subjected to irradiation, their readings vary, or they malfunction altogether. This underscores the importance of determining exposure temperature conditions from readings of thermostated subloop transducers located outside the channel active region.

The possible heat transfer conditions in irradiation are studied during the initial testing period in order to cope with this problem, and deviations in the readings of thermometer type sensors in the subloop from the readings of thermocouples placed on the specimens to be irradiated or the fuel assembly undergoing tests are also checked out. These deviations are studied to see how they vary with the absolute level of temperatures in the channel with the rate of coolant flow through the channel, and with various other factors. The dependences found by this method are continually monitored throughout the course of the experiment, thereby making it possible to continue the testing process even when the thermocouples are not functioning properly.

In this case, the temperature of the fuel-element surface can also be checked by calculations, using the results of neutron physics measurements of power output over the cross section and over the height of the specimens. Once the thermal output power of the channel (or fuel assembly), the flowrate and temperature of the coolant at entry to the fuel assembly, and neutron measurements data are known, there is little problem in obtaining the temperature of the cladding at any point on the fuel assembly from the available computational formulas.

Protection of Fuel Assemblies to be Tested. The following methods for protecting against power excursions are in use in subloop testing work: using readings of thermometer transducers at the exit from channels, as corrected according to readings of thermocouples placed on the fuel assembly in the inital exposure period; using readings of thermocouples at the exit from the fuel assemblies (when their readings are deemed reliable), generally in the first period of the experiment; using readings of the instrument monitoring the temperature drop across the fuel assembly (when the rate of coolant flow through the channel is being monitored simultaneously).

Note that these methods may prove to be inapplicable to channels of BWSL subloops working in the boiling mode. Protection of these channels against power excursions is therefore carried out indirectly by means of the working channel in the MIR reactor, which is located in the moderator slug adjacent to the channels.

In the nonboiling mode, when the power output of the subloop channel can be determined with ease, the relationship between the power outputs of the test channel and the working channel, selected for the purposes of emergency protection, is found. In the boiling mode, emergency protection goes into effect when the power setpoint of the working channel, as recorded by the appropriate instruments, is exceeded.

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RESEARCH AND DEVELOPMENT WORK ON PEARLITIC STEELS FOR STEAM GENERATORS IN SODIUM-COOLED-REACTOR NUCLEAR POWER PLANTS

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One of the salient features of the service of materials used in steam generators for nuclear power plants with sodium-cooled reactors is the fact that they must contact liquid sodium at temperatures to 550°C, on the one hand, and water (or steam), on the other hand [1, 2].

The need for corrosion stability on sodium streams naturally dictates that the steam generator use austenitic chromium – nickel steels such as used for the remaining elements of the first and second subloops of the reactor, but these materials are ruled out because of the danger of corrosion cracking the steam and water environment. The use of high-nickel austenitic steels is ill-advised because these grades of steel present such shortcomings as poor machinability and recalcitrance to advanced production techniques in the fabrication of semifinished products, poor weldability, high cost, etc. It would be preferable to rely on thermostable steels of the pearlitic class, particularly since a large amount of experience has been accumulated on applications of these grades of steel in boilers used in conventional electric power generating plants. The most important characteristic favoring the selection of specific compositions of these steel grades must be their corrosion stability in liquid sodium streams.

The effect of sodium as a decarburizing medium, studied widely at the present time in the USSR and in other countries, rules out the use of insufficiently alloyed pearlitic steels for service at high temperatures in contact with austenitic steels. Otherwise, we might have to be content with inadmissible losses in the strength of the pearlitic steel and embrittlement (as a result of carburization) of the austenitic steel. When selecting alloying of steel for steam generators, we have to consider the fact that increasing the quantity of carbide-forming elements in the steel will inevitably lead to impairment of weldability. Certain difficulties accompanying the selection of high-alloy thermostable steels may also be encountered in metallurgical plants in the fabrication of semifinished products, especially tubes and large forgings.

The principal problem to be tackled in developing materials for steam generators actually revolves around the selection of some steel of the pearlitic class alloyed with an amount of carbide-forming elements such that the required resistance to decarburization in sodium will be attained and at the same time severe impairment of machinability and adaptability to mass production techniques as compared to unalloyed steel.

TABLE 1. Proportionality Factors K_E and Free Energy of Formation of Carbides of Different Elements [6-8]

Characteristics of bonding of alloying elements to carbon	Alloying elements									
	Ti	Nb	V	Cr	Mo	Mn	Al	Fe	Co	Ni
Proportionality factor K_E	−41	—	−25	−13,2	—	−4,15	−1,5	0	+2,3	+4,2
Free energy of formation of carbides A_i at 600°C, kcal/mole	−42	−33	−27	−13,5	−7,0	−4,2	—	1,0	+2,5	+6,8

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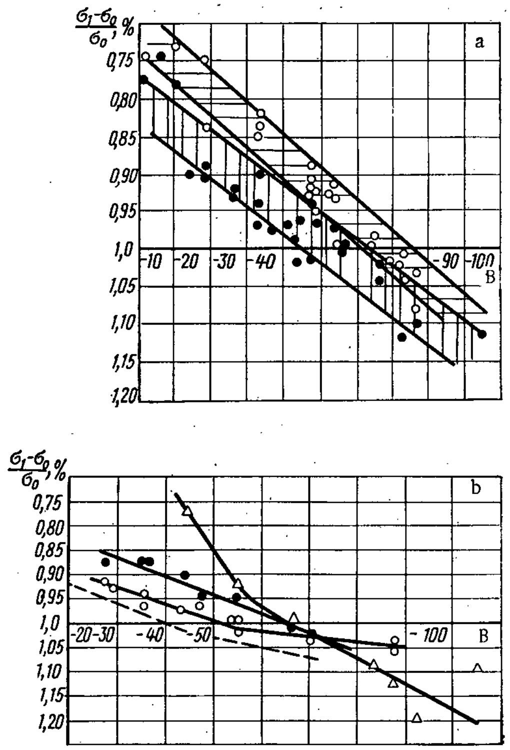

Fig. 1. Dependence of relative change in tensile strength of specimens exposed to sodium stream and to argon stream on alloying level B, in capsule tests (a) and in sodium stream (b): a: ●) 600°C, 4000 h; ○) 650°C, 4000 h; b: △) 650°C, 3000 h; ●) 600°C, 4000 h; ○) 550°C, 4000 h; ----) 450°C, 4000 h.

TABLE 2. Chemical Composition of Heats Investigated (wt. %)

Number of heats	C	Cr	Mo	V	Nb	Ti	W
1—5	0,05/0,12	1,27/4,78	0,27/0,79	—	—	—	—
6—15	0,05/0,10	1,07/4,50	0,34/0,80	0,25/0,70	—	—	—
16—26	0,04/0,12	2,00/4,50	0,60/1,00	0,25/0,45	0,17/0,85	—	0/1,39
27—31	0,08/0,12	1,50/2,50	0,50/1,00	—	—	0,25/0,65	—
32—33	0,08/0,12	2,00/2,50	0,80/1,00	—	0,30/0,60	0,30/0,50	—
34	0,14	2,38	0,89	0,32	0,36	0,49	
35—38	0,07	2,00/2,50	0,80/1,00	—	0,50/1,00	—	—

Transfer of carbon from pearlitic temperature-resistant steels to austenitic steels via liquid sodium is ascribed to the different thermodynamic activity of carbon [3, 4], while the rate of the transfer process is determined by the diffusion coefficients in the contacting steels and in the sodium [3-5]. A decrease in the amount of carbon transferred can be achieved either by reducing the diffusion coefficients of carbon in pearlitic steels or by lowering carbon activity in pearlitic steel to the level of carbon activity in austenitic steel. This can be brought about by lowering the carbon content and by introducing carbide-forming elements (Ti, Nb, V, Cr, Mo) into the steel, since these elements simultaneously lower the thermodynamic acitivity of carbon and lower its diffusion coefficient. The activity coefficient of carbon f_C in austenite simultaneously alloyed with different elements E_1, E_2, E_3 is found from the equation [5, 6]

$$f_C = f_C^{Fe} f_C^{E_1} f_C^{E_2} f_C^{E_3}, \tag{1}$$

where f_C^E is the activity coefficient of carbon when a particular alloying element is introduced into the unalloyed austenite.

There also exists a quantitative dependence of f_C^E on the content of the alloying element, given by the equation [6]:

$$\ln f_C^E = K_E N_E, \tag{2}$$

where N_E is the content of the alloying element; K_E is a proportionality factor which provides a qualitative characterization of the alloying elements.

Proportionality factors K_E [6, 7] calculated on the basis of experimental data reported by many different authors (Table 1) have been compiled for some of the elements. The lowest K_E value is featured by titanium, as evidence of its great influence in diminishing carbon activity in steel.

Similar effects exerted by alloying elements have been found in the case of pearlitic steels as well [9]. It has been shown that carbon solubility in alloyed ferritic steel, a determining factor of its activity (with loss of solubility meaning loss of activity), is found as a function of the ratio of the content of the alloying element to the content of carbon (Me/C).

The diffusion coefficient of carbon in metal, when various elements are introduced, varies in exactly the same manner [10, 11]. Elements for which the K_E value is of negative sign (see Table 1) exhibit low diffusion coefficients for carbon in steel. The lowest value occurs when titanium is introduced, while, when nickel and silicon, which have positive K_E values, are introduced, the diffusion coefficient of the carbon rises.

Another qualitative characteristic of carbon bonding to the alloying elements is the free energy of formation of the carbides. Introduction of an element having a low energy of formation of its chemical compound with carbon lowers carbon activity in the steel and at the same time lowers the diffusion coefficient of carbon in the steel.

Comparison of the proportionality factors K_E and the values of the free energy of formation of the carbides A_i shows that these are commensurate in qualitative terms. This means that we can make use of the A_i values instead of the K_E values, while taking the carbon content in molar fractions in the other elements into account, as well as the stoichiometric ratio of carbon and the alloying element in the formation of the carbide, to arrive at a qualitative assessment of the steels in terms of their resistance to decarburization; this is aided by introduction of the parameter B characterizing the alloying level and taking into account the ability of the steel to resist decarburization.

The parameter B is given by the equation

$$B = \sum_{1}^{i} \frac{A_i}{k} \frac{\mathrm{Me}_i}{c}, \tag{3}$$

where c is the carbon content in mole fractions; k is a coefficient taking the stoichiometric ratio of carbon and the alloying element of the corresponding carbide into account.

Corrosion tests were carried out on a wide array of compositions of pearlitic steels in which the content of the alloying elements was allowed to vary over rather wide ranges, in order to confirm the suggested regularity, and to make a study of the effect of alloying of the steel on the amount of carbon transferred (see Table 2).

The tests were carried out on different arrangements: in capsules (including use of the radioactive isotope C^{14}), in subloops with convective and forced circulation of liquid sodium at flowspeeds v up to 4 m/sec, maximum subloop temperature 450-650°C, and temperature drop 150-220°C. The purity of the sodium, in terms of oxygen impurities, was usually less than that assumed to prevail under real conditions (0.05 to 0.008 wt.% in our experiments). In subloops with convective and forced circulation, the area ratio of the surface of austenitic and pearlitic steels was greater than 200, but much less ($S_{aus}/S_{pe} \approx 2$) in the sodium subloops of nuclear reactors. Hence, the testing conditions for the pearlitic steels were much more stringent than the real conditions under which the materials see service in steam generators.

An estimate of the results of these tests was made in terms of changes in mechanical properties, in carbon content, and in the microstructure of flat and cylindrical specimens. Repeated tests established the possibility of using B as a parameter characterizing the resistance of the steel to decarburization. This was confirmed by capsule tests with sodium, at oxygen contents ~0.05 wt. % at 600° and 650°C for 4000 h (Fig. 1a). The criterion for decarburization was the relative change in the tensile strength of specimens exposed to the sodium stream (σ_1) compared to specimens exposed to an argon stream (σ_0).

Preliminary experiments were set up to determine how the carbon content affects mechanical properties, and showed that a drop of 0.01 wt. % in carbon content lowers tensile strength by 1.5 to 2 kg_{force}/mm^2. This gives some idea of the carbon content. A determination of carbon content in failed specimens also confirms the regularity in question. We can see in Fig. 1a that absence of decarburization at 600°C ($\sigma_1 - \sigma_0/\sigma_0 = 1.0$) has been detected in chromium – molybdenum steels alloyed with 1 wt. % niobium or 0.5 wt. % titanium, and also in steels containing more than 4 wt. % chromium and 0.3 wt. % vanadium. The results cited are in excellent agreement with the results reported by other investigators [3, 5]. The parameter B characterizing these steels is –60 to –70 kcal/mole when the testing temperature is 600°C, and –75 to –80 kcal/mole when the testing temperature is 650°C. At lower temperatures, less highly alloyed steels may prove sufficiently stable against decarburization.

Tests were conducted on several pearlitic steels in forced sodium flow (v ≈ 3 m/sec) with an oxygen content ~0.006 to 0.008 wt. % at temperatures of 450°, 550°, 600°, and 650°C. Analysis of the test results (Fig. 1b) suggests that distinct chromium – molybdenum steels alloyed with different elements are serviceable at temperatures from 450° to 550°C. The parameter B should be not less than –40 kcal/mole at temperatures up to 450°C, for the particular steel, in order to ensure high resistance to decarburization on the part of the steel, and not less than –50 kcal/mole at temperatures to 550°C.

Theoretical analysis (based on the parameter B), and results of comparative tests run in capsules, in convective subloops, and in forced-flow subloops, showed that steels with minimum carbon content must

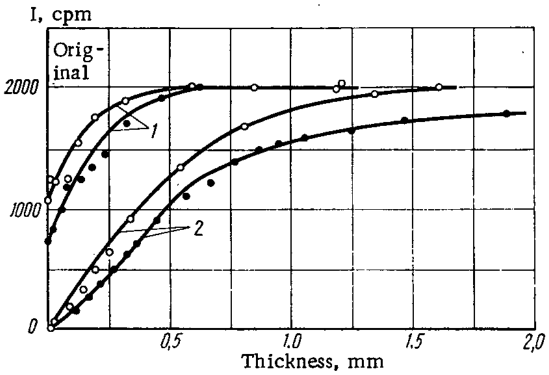

Fig. 2. Depth and degree of decarburization of steels 1Kh2MFB (curves 1) and 1Kh2M (curve 2) exposed to sodium stream at 600°C for 2000 h (○) and 5000 h (●).

TABLE 3. Chemical Composition of Steel Grades Recommended (in wt. %)

Grade of steel	C	Cr	Mo	V	Nb
1Kh2M	0,08 / 0,12	2,00 / 2,50	0,60 / 0,80	—	—
1Kh2MFB	0,08 / 0,12	2,00 / 2,50	0,80 / 1,00	0,25 / 0,35	0,55 / 0,45

be used. However, when we recall that fabrication of low-carbon steels runs into certain difficulties, and lowering the carbon content may also lower the strength properties of the material, while an increase in carbon content will contribute to inferior weldability, we find that the optimum carbon content must lie somewhere in the range of 0.1 wt. %. The goal of high resistance to decarburization is best met by a steel containing 0.1 wt. % carbon up to temperature of 450°C, and in our view not less than 2.0 wt. % chromium.

Introduction of up to 1.0 wt. % molybdenum is dictated by the requirements of high refractoriness and structural stability, and the molybdenum does the added service of enhancing resistance to decarburization, by lowering the diffusion coefficient of carbon in ferrite.

At still higher temperatures (up to 550°C), it will not be possible to attain high resistance to decarburization without resorting to strong carbide-forming elements (titanium, niobium, vanadium), but introduction of these elements inevitably entails greater difficulties in adapting the resulting steels to production techniques. For example, introduction of 0.5 wt. % titanium and 0.5 wt. % niobium into chromium – molybdenum steel (see heat No. 33 in Table 2), or introduction of niobium, vanadium, and titanium (see heat No. 34, Table 2), brought about a drastic loss of impact strength (~1 $kg_{force} \cdot m/cm^2$) and impairment in weldability. A prime example of poor weldability is provided by the steel grade ÉI-631, which contains up to 1 wt. % niobium in its composition.

The above discussion argues in favor of the use of chromium – molybdenum steel, with additions of vanadium and niobium in slight quantities, for steam generators whose operating temperatures are upwards of 450°C. Introduction of 0.3 wt. % vanadium and 0.3 wt. % niobium into chromium – molybdenum steel will decrease decarburization in sodium appreciably, but this is paid for with greater difficulties in welding and machining the steel.

We see that, in the case of steam generators designed for service at temperatures to 450°C, 1Kh2M steel was recommended, while 1Kh2MFB steel was recommended for steam generators designed for service at temperatures to 550°C (see Table 3). The steels selected were subjected to further corrosion tests in different types of sodium-stream installations. These tests provided further confirmation of the excellent resistance to decarburization presented by the steels 1Kh2M and 1Kh2MFB.

In making a selection of steels for steam generators which would lead themselves best to manufacturing processes, minimization of decarburization of the steels in a sodium stream was kept in mind as the goal. The method of labeled atoms (C^{14}) was brought to bear in studying the decarburization kinetics, in determining the effective diffusion coefficients of carbon in these steels, and in finding the possible depths and degrees of decarburization.

Figure 2 displays results of one of the numerous tests conducted on the steels 1Kh2M and 1Kh2MFB in capsules with commercial grade sodium (oxygen content 0.05 wt. %). The depth and degree of decarburization was determined by the method of layer-by-analysis of the intensity of β-radiation from specimens exposed in capsules of austenitic steel. We realize from Fig. 2 that introduction of 0.3 wt. % niobium and 0.3 wt. % vanadium (curve 2) drastically reduces both the depth and the degree of decarburization. Since the effective diffusion coefficients of carbon in 1Kh2M steel at 450°C and in 1Kh2MFB steel at 550°C are roughly identical, the depth of decarburization occurring over the time the steam generators are on stream will be approximately 0.25 mm for 50% carbon losses and 0.8 mm for 25% carbon losses.

The radioactive isotope C^{14} was successfully employed as a tracer in establishing the degree of carburization of austenitic steel Kh18N9 employed in the vessels of heat exchangers. Calculations backed up by experiments show that when the amount of carbon transferred from the pearlitic steels 1Kh2M and 1Kh2MFB to the austenitic steel Kh18N9 remains constant, and when the ratio of surface area presented by pearlitic and austenitic steels in real power plants prevails, the maximum carbon content on the surface of the austenite will not be greater than 0.30 to 0.40%.

Consequences of carbon transfer as it affects the mechanical properties of the pearlitic steels and austenitic steel were evaluated at the same time. This was done by making laboratory heats of those grades of steel with reduced carbon content (to 0.04 wt. %) in the pearlitic grades and enhanced carbon content (to 1 wt. %) in the austenitic grade. The strength characteristics of the pearlitic steels (long-term ultimate strength and yield point) were lowered by about 10-15% when the carbon content was reduced to the extent likely to happen through decarburization in a sodium stream. The long-term strength and the resistance to thermal fatigue exhibited by Kh18N9 austenitic steel with enhanced carbon content (0.3 to 0.4 wt. %) remained on the same level as that of steel containing 0.08 to 0.10 wt. % carbon. The observed fall-off in long-term ductility does not lead to any premature failure. Accordingly, carbon transfer from 1Kh2M and 1Kh2MFB pearlitic steels to Kh18N9 austenitic steel does not exert any severe effect on the serviceability of steam generators and heat exchangers made from these materials.

Investigations of the mechanical properties of both steel grades were carried out over a wide range of temperatures and times, and studies were made of the resistance to thermal fatigue, long-term strength characteristics, optimum heat treatment conditions, and other factors. Both grades of steel respond satisfactorily to welding. The high corrosion resistance of chromium – molybdenum steels in a water – steam medium has been generally acknowledged, since they have been in use over a protracted period in steam boiler structures of electrical power plants under high-temperature conditions, and the steel grades 12Kh-1MF, 15Kh1MF, and ÉI-531, close to them in chemical composition, have also received excellent recommendations for this type of high-temperature service. Laboratory tests in water and steam confirmed the excellent resistance presented by these steels to corrosion in water (or steam).

The steel 1Kh2M was therefore recommended and accepted for steam generators in nuclear power plants with sodium-cooled reactors for service at temperatures up to 450°C, and the steel 1Kh2MFB was recommended for service at temperatures to 550°C.

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ESTIMATION OF THE DURABILITY OF AUSTENITIC STEELS FOR NUCLEAR POWER STATIONS WITH SODIUM COOLANT

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The construction of large nuclear power stations in the USSR with sodium as coolant has involved the solution of various technical problems of design [1-3]. One serious problem was to develop structural materials and to estimate their service lives. In this article we discuss the most important results of this research on the service lives of steels for the main elements of the first and second circulation loops and for certain components inside the reactor.

Operating Conditions of Materials. The maximum temperature of the liquid sodium in plants of various designs varied between 500 and 650°C.

With sodium, as in the cases of other liquid metal coolants, the internal pressure is comparatively low. Thus the mechanical stresses in the reactor vessel, piping, and other components are also low. On the other hand, the good heat-transfer capacity of the liquid metal causes a high level of thermal stresses, both steady and cyclic, due to the sharp temperature changes. The sizes of these stresses and the number of thermal cycles in plant of this type are greater than in many other power plants. Consequently, thermal fatigue is a very important factor and must be allowed for in estimating the strength of the structure.

Another type of load in these plants is due to reactive forces caused by incomplete compensation of the thermal expansion of the piping, or in welded rigid components. Experience of the operation of thermal power stations in the USSR and abroad has revealed that cracks may occur in welded joints in steam pipes made of austenitic steels stabilized with niobium or titanium. This type of damage, known as "local failure," is certainly impermissible in plant of the type in question.

In addition to the above requirements, the steels also need high corrosion resistance and low embrittlement under the action of heat and radiation.

Finally, the steels must be easy to work in metallurgical production conditions and must be weldable.

Main Characteristics of Steels. Steels for the above purpose were chosen on the basis of laboratory studies of austenitic steels of the simplest compositions.* As a result of this research the steels recommended were Kh18N9 and Kh16N11M3. The chemical compositions of these steels are given in Table 1.

The minimum yield points were guaranteed by the manufacturers. For both steels at room temperature $\sigma_{0.2} \geq 20$ kg/mm^2; at 530°C, $\sigma_{0.2} \geq 10$-12 kg/mm^2 (steel Kh18N9); at 600°C $\sigma_{0.2} \geq 10$ kg/mm^2 (steel Kh16N11M3). Steel Kh16N11M3 has some advantage in high-temperature strength over steel Kh18N9. Thus for the former, the stress-rupture strength after 10^5 h at 650°C is 5.5 kg/mm^2, whereas for the latter it is 7.0 kg/mm^2.

Extensive industrial tests on forgings, sheets, and tubes reveals that both steels are metallurgically easy to work and easy to weld.

Steel Kh18N9 for use in power stations has a composition close to the standard steel of this mark widely used in the USSR. However, it has a limited carbon content (0.1% as against 0.12%). The titanium content is also kept down (0.1%), though it must be added to give a stable fine-grained structure.

* Perlite steels are unsuitable owing to their low corrosion resistance and because of difficulties of welding and heat treatment.

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TABLE 1. Basic Chemical Compositions of Steels Kh18N9 and Kh16N11M3

Steel	Contents of elements, %				
	C	Cr	Ni	Mo	Ti
Kh18N9	⩽0,1	17—19	8—10	—	⩽0,1
Kh16N11M3	⩽0,08	15—17	10—12	2,0—2,5	⩽0,1

In addition, for both steels the following specifications are laid down:

1) The steel is to be prepared from fresh specially-pure charges, so as to get a high degree of freedom from harmful impurities.

2) The content of ferrite phase is to be strictly regulated (1-4%) to improve the weldability and avoid thermal embrittlement.

3) Semifinished products are to be 100% ultrasound-tested to avoid turning out components with flaking, cracks, or other defects.

Resistance to Thermal Fatigue. As stated above, operation of nuclear power stations with liquid metal coolants involves high thermal stresses, especially during nonsteady temperature conditions. A long time elapses between successive thermal cycles, and during this time individual components are acted on by gradually relaxing thermal stresses.

In starting our research we were confronted with two alternative approaches to the selection of a structural material. The first consists in formal summation of the mechanical and thermal stresses and adoption of a reserve margin of the strength characteristics. In this case, in the design of the plants we should require steel with very good strength characteristics, which would be awkward from a technological viewpoint.

The second, more correct approach to the choice of material consists in dividing the loads and determining the requirements from the actual service conditions of the material. This approach involves the necessity of establishing some minimal strength level (yield point, stress-rupture strength) which exceeds the actual mechanical loads with a certain margin, and some thermal fatigue resistance which will be the theoretical characteristic. A preliminary analysis revealed that with this approach one can correctly choose austenitic steels of simple composition, which are easy to work and weld.

Realization of the approach (the only correct one) involved some difficulties. Despite the extensive research which has been done on thermal fatigue, many problems, even some general ones, remained obscure. Thus there was some disagreement on the basic parameters which must be assigned in comparing different materials; practically no work had been done on the time dependence of the thermal fatigue; and so forth. Thus there were no unique methods of investigation.

Extensive research was performed to fill these gaps [4].

As the basic parameters most correctly representing the behavior of the materials, we chose $\Delta\varepsilon$, the total deformation per cycle (elastic plus plastic), t_{max}, the maximum temperature of the cycle, and τ_c, the mean time spent at the maximum temperature in one cycle.

To study the time dependence of the thermal fatigue resistance we used the well-known Coffin apparatus. An important departure from Coffin's original method was that the tubular specimens were periodically kept for a given time at the maximum temperature of a cycle. During this process the specimens were acted on by relaxing thermal stresses, the level of which was restored after successive cooling and heating.

Experimentally we plotted two main graphs, serving as a basis for thermal fatigue calculations: we plotted the number of thermal cycles to failure versus the total deformation per cycle, and also the number of cycles to failure versus the retention time. By processing the graphs mathematically one can obtain the expression

$$N=\frac{1}{4(\tau_c)^b}\left(\frac{A}{\Delta\varepsilon}\right)^{\frac{1}{a}},$$

where A, a, and b are coefficients which are found experimentally for each mark of steel from the deformation and time dependences.

This formula is the fundamental equation for calculations on the thermal fatigue of structural elements. The values of $\Delta\varepsilon$ at the danger points are found by simultaneously solving problems in the theory of thermal conduction and in the theory of elasticity.* To avoid errors associated with the transition from laboratory

*If the thermal stresses exceed the elastic limit, the values of $\Delta\varepsilon$ found from the formula of the theory of elasticity must be corrected by introducing appropriate factors.

TABLE 2. Influence of Austenitization on Tendency of Steels Kh18N9 and Kh18N10T to LFZS (tested at 650°C for 10 h)

Steel	State of specimen	Number of cycles	Results of luminescent monitoring	Depth of cracks, mm
Kh18N9	Initial	5	Narrow cracks	0.57
Kh18N9	After austenitization	14	Slight tears	0.175
Kh18N10T	The same	3	Continuous cracks	4.3
Kh18N10T	" "	5	The same	9.0

specimens to full-scale components, the thermal fatigue calculations must allow a certain margin. From analysis of published data and from special experiments on the effects of certain factors (scatter of experimental data, scale factor, surface cleanness, effect of rigidity of state of stress, influence of the medium, influence of loss of plasticity owing to irradiation, etc.) it was deemed possible to take the safety margin in the number of cycles to be 20.

The permissible cyclic deformations for operation of plant with a service life of the order of 200,000 h are as follows: 0.45% for working temperatures up to 550°C; 0.34% for working temperatures up to 600°C, and 0.27% for working temperatures up to 650°C.

If several types of cyclic load act on a component, then we must base our approach on the concept of independent action of the loads and summation of the damage [4, 5]. A similar approach can be used to allow for the simultaneous action of thermal and mechanical stresses.

In conclusion we note that the calculations have revealed that the thermal and mechanical loads arising in practice are safe for components made of steels Kh18N9 and Kh16N11M3.

Tendency to Local Failure in Zones near Seams. As a result of analysis of the methods of studying the resistance to local failure in zones near seams (LFZS), it was established that they have certain drawbacks which render them unsuitable for assessing the service lives of welded joints in nuclear power plants.

A new test was therefore developed, based on a welded T-piece made from bars 40 mm thick. Such a specimen permits creation of high stresses in the corner seams by welding loading beads on to the base of the T-piece. We monitored the cracks arising near the corner seams after the specimens had been kept for a long time at various temperatures. In addition to these tests with once-only welding of the beads and once-only heating, a more sensitive method was also used in which the seam zone was subjected to cyclic loads. In this case the specimen of austenitic steel was kept for 10 h in a furnace at 650°C. After cooling in air, an additional bead was welded on and the specimen again kept at 650°C for 10 h. After three such operations the bead is removed, a new layer welded on, and so on until a crack forms.

In the experiments described below, as well as steels Kh18N9 and Kh16N11M3 we investigated titanium-stabilized steel Kh18N10T,* which can be regarded as representative of steels with marked tendency to LFZS. To account for the possible influence of the metallurgical smelting conditions, T-pieces were made from steels Kh18N9 and Kh16N11M3 from ten different commercial smeltings. The specimens were kept at 650°C for up to 1000 h. In no case did we observe the formation of cracks. In specimens of steel Kh18N10T cracks arose after 30-50 h at 650°C. At 550°C cracks were observed after about 1000 h. Thus it was shown that steels Kh18N9 and Kh16N11M3 are much less liable to LFZS than steel Kh18N10T. This result must in the first place be attributed to the presence in steel Kh18N10T of a comparatively high titanium content. The high metallurgical quality of the steels Kh18N9 and Kh16N11M3 also played an important part.

For a more detailed comparison of the two steels with small tendency to LFZS we carried out experiments on cyclic heating and cooling of T-pieces. These experiments revealed that steel Kh16N11M3 is somewhat less prone to LFZS than steel Kh18N9.

*Titanium content 0.5-0.7%.

It is recommended that welded joints of austenitic steels stabilized with titanium should be subjected to austenitization. There is disagreement on the necessity of austenitizing welded structures made of steel which is not very liable to local failure. This is largely due to the lack of a method for assessing the effects of austenitization on the tendencies of these steels to LFZS. The results of experiments with T-piece specimens are listed in Table 2.

The data in Table 2 again confirm the marked advantage of steel Kh18N9 over steel Kh18N10T. They also enable us to draw conclusions on the advantageous influence of austenitization on steel with little tendency to LFZS.

Thus in building structures for nuclear power stations of steels Kh18N9 and Kh16N11M3, the danger of LFZS is minimized, especially if austenization is used.

In estimating the service life of the steels, other experiments were also performed which, in particular, revealed that at the working temperatures steels Kh16N11M3 and Kh18N9 have high corrosion resistance to liquid sodium, does not undergo thermal embrittlement, and its properties are relatively unaffected by irradiation.

From the above investigations we can conclude that steels Kh18N9 and Kh16N11M3 should prove reliable in the structures of nuclear power plants with sodium coolant.

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ESTIMATE OF THE EFFICIENCY OF FUEL ELEMENT JACKETS IN FAST REACTORS

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In designing fast reactors (USSR), it became necessary to devise methods for calculating the efficiency of airtight cylindrical fuel elements. Estimates of the efficiency of such fuel elements are considered for different models of simultaneous operation of the fuel and the jacket [1].

According to the first model, a clearance, filled with gas or the sublayer material, which remains liquid throughout the range of operating temperatures (for instance, sodium), is maintained between the fuel and the jacket throughout the operating life of the fuel element. In this case, the jacket experiences, besides the external coolant pressure, which increases with fuel depletion, also the internal gas (or liquid) pressure and the thermal stresses due to the thermal field gradients, which are subject to relaxation in time. Moreover, the jacket is subject to thermal cycling (as a result of cooling and transient operating conditions).

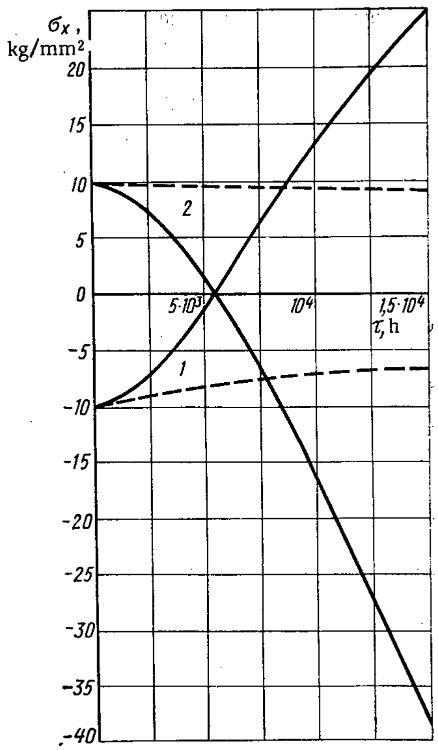

Fig. 1. Operating axial stresses in the hot (550°C, curve 1) and the cold (500°C, curve 2) sides of the jacket for the median section along the fuel element: ———) with an allowance for the nonuniform swelling of steel; - - - -) without an allowance for the nonuniform swelling of steel.

If the external coolant pressure is high (gas-cooled reactor), the jacket may lose its stability, which can cause the collapse of the unsupported compensating volume or produce a longitudinal crimp along a portion of the fuel element, where the oval jacket rests against the fuel core.

An irreversible increase in the jacket length due to thermal cycling (thermomechanical "ratchet" [2]) is also possible after an oval jacket comes into contact with the core.

In another model, it is assumed that the jacket and the core are in contact throughout the operation or during a part of it. Therefore, during the constant-load operation of the reactor, the jacket must withstand not only the internal pressure of the gaseous fission products, but also the pressure exerted by the swelling fuel.

If the fuel element is designed so that the core is rigidly bound to the jacket (for instance, through a solid, hard sublayer with a good bond at the contact surface), reversible plastic strain can occur in the jacket as a result of thermal cycling. Then, in determining the kinetics of the stressed state of the jacket, it is necessary to take into account the change in the stress – strain diagram of the material from one thermal cycle to another [3, 4]. In this case, the jacket operates under especially difficult conditions because of the strength requirements, which must be considered in designing the fuel elements (thus, it is necessary to match the thermal expansion coefficients of the fuel and the jacket). In principle, reversible plastic strain can also be caused by nonuniformities of the jacket's thermal field (in the case of considerable peripheral temperature nonuniformities and large thermal fluxes). It should be

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mentioned that tangential and radial temperature gradients do not cause reversible plastic strain in fuel elements designed for fast reactors.

Since stresses in the jacket vary to a large extent in operation, the strength calculations must be performed with respect to the cumulative damage by using the well-known concepts forwarded by Miner [5], which were developed further in [6-9].

Considering the damage to the jacket material due only to the prolonged action of stresses at high temperatures, we write the following condition for the time t_{op} of safe jacket operation:

$$\int_0^{t_{op}} \frac{(\sigma_e K_\sigma)^n \, dt}{D} = a_B, \tag{1}$$

where $a_B \approx 0.8$ is the factor of susceptibility to damage at rupture; K_σ is the relative stress safety factor; D(T) and n(T) are the long-term strength characteristics ($\delta_e^n t = D$), determined from tests inside the reactor; and δ_e is the equivalent stress [8] responsible for mixed failure (along the boundaries and through the body of grains).

Another possible approach is to use the intensity of the inelastic strains accumulated in the material as the criterion in estimating the efficiency of a jacket:

$$\varepsilon_i = \frac{\sqrt{2}}{3} \sqrt{(\varepsilon_1 - \varepsilon_2)^2 + (\varepsilon_2 - \varepsilon_3)^2 + (\varepsilon_3 - \varepsilon_1)^2} \leqslant \frac{\varepsilon_{iB}}{K_\varepsilon}, \tag{2}$$

where ε_{iB} is the rupture strain, determined from tests inside the reactor, and $K\varepsilon$ is the strain safety factor.

The strain criterion can be conveniently used for relatively low temperatures, when long-term strength calculations are not representative.

Effect of the Peripheral Nonuniformity of Temperature Distribution on the Swelling of Steel

Nonuniformities in the peripheral temperature distribution, which are especially pronounced for fuel elements near the walls of in corners, can produce considerable stresses. Therefore, in certain cases, it is absolutely necessary to take into account their effect on the jacket efficiency. It should be mentioned that nonuniformities in the peripheral temperature distribution affect the fuel element efficiency especially in the case of large integral fast neutron fluxes, since they lead to nonuniform swelling of steel. Nonuniform swelling along the jacket perimeter under conditions of confined fuel element deformation (small clearances) causes additional stresses and mechanical deformations in the jacket.

The azimuthal development of the temperature fields of fuel elements has been investigated in many experimental (casting of mock-ups of fuel element packages) and theoretical papers [10-12].

The flexure of fuel elements due to tolerances usually amounts to only a fraction of the thermal bending of a free jacket, so that the element axis can be considered as being virtually straight. In view of the small axial temperature gradients, individual jacket sections can be considered independently of each other, thus reducing the problem to a plane problem.

The method of variable elasticity parameters, described in Birger's papers [13, 14], is used for determining the kinetics of the stressed – strained state of jackets under conditions of nonisothermic loading and irradiation with an allowance for instantaneous plastic and creep deformations.

The change in the stress – strain diagram of the jacket material caused by irradiation is taken into account. The problem consists in considering an anisotropic elastic jacket with variable elasticity parameters and additional strain; the jacket temperature is determined by the function T(z, Θ, t).

The loading process is divided (in time) into a number of small stages, and the increments of the stress and strain components are determined for each stage.

For a thin-walled jacket ($\sigma_r = 0$), the physical dependences (in terms of increments)* are written for the n-th loading step ($\Delta_n t = t_n - t_{n-1}$):

* The notation is explained in [13].

$$\Delta_n\varepsilon_x = \langle C_{11(n_i)}\rangle \Delta_n\sigma_x + \langle C_{12(n)}\rangle \Delta_n\sigma_\Theta + \langle \varphi_{xT(n)}\rangle \Delta_n T + \langle \varphi_{x\eta(n_i)}\rangle \Delta_n\eta + \langle \varphi_{xc(n_i)}\rangle \Delta_n t + \frac{1}{3}\Delta_n S;$$
$$\Delta_n\varepsilon_\Theta = \langle C_{12(n_i)}\rangle \Delta_n\sigma_x + \langle C_{22(n_i)}\rangle \Delta_n\sigma_\Theta + \langle \varphi_{\Theta T(n)}\rangle \Delta_n T + \langle \varphi_{\Theta\eta(n_i)}\rangle \Delta_n\eta + \langle \varphi_{\Theta c(n_i)}\rangle \Delta_n t + \frac{1}{3}\Delta_n S, \tag{3}$$

where the elasticity parameters are given by

$$C_{11} = \frac{1}{E} + (2\sigma_x - \sigma_\Theta)^2 \frac{F_\sigma}{6\sigma_i};$$
$$C_{22} = \frac{1}{E} + (2\sigma_\Theta - \sigma_x)^2 \frac{F_\sigma}{6\sigma_i}; \tag{4}$$
$$C_{12} = -\frac{\mu}{E} + (2\sigma_x - \sigma_\Theta)(2\sigma_\Theta - \sigma_x)\frac{F_\sigma}{6\sigma_i},$$

while the functions of the additional strains are given by the following expressions:

$$\varphi_{xT} = \frac{d(\alpha T)}{dT} - \frac{1}{E^2}\cdot\frac{dE}{dT}(\sigma_x - \mu\sigma_\Theta) - \frac{1}{E}\cdot\frac{d\mu}{dT}\sigma_\Theta + \frac{2\sigma_x - \sigma_\Theta}{3}F_T;$$
$$\varphi_{\Theta T} = \frac{d(\alpha T)}{dT} - \frac{1}{E^2}\cdot\frac{dE}{dT}(\sigma_\Theta - \mu\sigma_x) - \frac{1}{E}\cdot\frac{d\mu}{dT}\sigma_x + \frac{2\sigma_\Theta - \sigma_x}{3}F_T;$$
$$\varphi_{x\eta} = \frac{2\sigma_x - \sigma_\Theta}{3}F_\eta; \quad \varphi_{\Theta\eta} = \frac{2\sigma_\Theta - \sigma_x}{3}F_\eta; \tag{5}$$
$$\varphi_{xc} = \frac{2\sigma_x - \sigma_\Theta}{3}F_c; \quad \varphi_{\Theta c} = \frac{2\sigma_\Theta - \sigma_x}{3}F_c;$$

the thermal expansion coefficient α, the elasticity modulus E, and the Poisson coefficient μ are considered to be known functions of the temperature T; the increment of the volume of the jacket material due to swelling is equal to $\Delta_n S = S(t_n) - S(t_{n-1})$, where $S = A_S(\Phi t)^{m_S} e^{Q_S/T - Q_S^*/T^2}$ is the experimental dependence of the steel volume on the integral fast neutron flux Φt and the temperature [15].

The plasticity function for the active load is

$$F_\sigma(\sigma_i T_\eta) = \frac{3}{2\sigma_i}\left(\frac{1}{E_k} - \frac{1}{E}\right); \quad F_\eta(\sigma_i T_\eta) = \frac{3}{2\sigma_i}\gamma; \quad F_T(\sigma_i T_\eta) = \frac{3}{2\sigma_i}\left(\beta + \frac{1}{E^2}\cdot\frac{dE}{dT}\sigma_i\right);$$

the plasticity function for the load release and neutral strain [14] is

$$F_\sigma = F_T = F_\eta = 0. \tag{6}$$

The quantity $E_k(\sigma_i T_\eta)$ is the indirect modulus of the deformation curve $(\sigma_i - \varepsilon_{ip})$ for the assigned temperature and level of radiation damage (T = const, η = const, σ_i = var); ε_{ip} is the cumulative plastic strain at a certain time of loading. The level of radiation damage is determined by a certain characteristic η (for instance, the number of displaced atoms [16] or the integral neutron flux). The thermal pliability coefficient $\beta(\sigma_i T_\eta)$ of a material that has sustained a certain level of radiation damage is determined in constant-stress tests at variable temperatures (η = const, σ_i = const, T = var). The quantity $\gamma(\sigma_i T_\eta)$ is the radiation pliability of the material, which must be determined by means of tests inside the reactor at a constant stress and temperature (σ_i = const, T = const, η = var). For an approximate determination of E_k, β, and γ, it is sufficient to have a set of tensile stress – strain curves for different temperatures, plotted on the basis of short-term tests of jacket material specimens, which have been irradiated to certain levels of radiation damage at operating temperatures.

According to the theory of hardening, the creep function is

$$F_c = \frac{3}{2}\sigma_i^{m-1}B(T, \varepsilon_{ic}), \tag{7}$$

where

$$B(T, \varepsilon_{ic}) = B_1 e^{-\frac{Q}{RT}} e^{H(T, \varepsilon_{ic})},$$

$H(T, \varepsilon_{ic}) = [\varepsilon_{ic}/\varepsilon_{ic}^*] H(T)$ for $\varepsilon_{ic} < \varepsilon_{ic}^*$ – the stage of unsteady creep; $H(T, \varepsilon_{ic}) = H(T)$ for $\varepsilon_{ic} \geq \varepsilon_{ic}^*$ – the stage of steady creep: $\varepsilon_{ic} = \int d\varepsilon_{ic}$ is the cumulative creep flow; and $\varepsilon_{ic}^* = B_2 e^{-Q^*/RT}\sigma_i^{m^*}$ is the deformation in transition to the stage of steady creep.

On the basis of data from [17], considering that all the components with the exception of the axial displacement $u = x\varepsilon_{x0}$ must be independent of the coordinate x in the case of a straight jacket axis, we write the differential equation for an elastic anisotropic thin cylindrical jacket in the following form:

$$\frac{1}{r^2}\cdot\frac{d^2}{d\Theta^2}\left[f_4\Delta_n\varepsilon_{x0}+\frac{f_{5\Theta}}{r^2}\left(\frac{d^2\Delta_n w}{d\Theta^2}+\frac{d\Delta_n v}{d\Theta}\right)\right]-\frac{1}{r}\left[f_2\Delta_n\varepsilon_{x0}+\frac{f_{1\Theta}}{r}\left(\frac{d\Delta_n v}{d\Theta}-\Delta_n w\right)\right]$$

$$=-\Delta_n p+\frac{1}{r^2}\cdot\frac{d^2(\Delta_n T_{2\Theta})}{d\Theta^2}-\frac{1}{r}\Delta_n T_{1\Theta}; \tag{8}$$

$$\frac{1}{r^2}\cdot\frac{d}{d\Theta}\left[f_4\Delta_n\varepsilon_{x0}+\frac{f_{5\Theta}}{r^2}\left(\frac{d^2\Delta_n w}{d\Theta^2}+\frac{d\Delta_n v}{d\Theta}\right)\right]+\frac{1}{r}\cdot\frac{d}{d\Theta}\left[f_2\Delta_n\varepsilon_{x0}+\frac{f_{1\Theta}}{r}\left(\frac{d\Delta_n v}{d\Theta}-\Delta_n w\right)\right]=\frac{1}{r^2}\cdot\frac{d(\Delta_n T_{2\Theta})}{d\Theta}+\frac{1}{r}\frac{d(\Delta_n T_{1\Theta})}{d\Theta}, \tag{9}$$

where $\Delta_n v$ and $\Delta_n w$ are the increments of the tangential and radial displacements; r is the jacket radius, and $\Delta_n p$ is the gauge pressure increment. The state of the basic surface of the jacket is determined by the equation $\int_{-\delta_1}^{\delta_2} E^*\mu_\Theta z\,dz = 0$. The following coefficients are used in Eqs. (8) and (9):

$$f_{1\Theta}=\int_{-\delta_1}^{\delta_2}E^*\mu_\Theta\,dz;\quad f_2=\int_{-\delta_1}^{\delta_2}E^*\,dz;$$

$$f_4=\int_{-\delta_1}^{\delta_2}E^*z\,dz;\quad f_{5\Theta}=\int_{-\delta_1}^{\delta_2}E^*\mu_\Theta z^2\,dz;$$

$$\Delta_n T_{1\Theta}=\int_{-\delta_1}^{\delta_2}\Delta_n\sigma_\Theta^*\,dz;\quad \Delta_n T_{2\Theta}=\int_{-\delta_1}^{\delta_2}\Delta_n\sigma_x^*\,z\,dz; \tag{10}$$

$$E^*=\frac{C_{12}}{C_{12}^2-C_{11}C_{22}};\quad \mu_\Theta=\frac{C_{11}}{C_{12}};\quad \mu_x=\frac{C_{22}}{C_{12}};$$

$$\Delta\sigma_x^*=E^*(\mu_x\Delta\varepsilon_x^*-\Delta\varepsilon_\Theta^*);\ \Delta\sigma_\Theta^*=E^*(\Delta\varepsilon_x^*-\mu_\Theta\Delta\varepsilon_\Theta^*);$$

$$\Delta\varepsilon_j^*=\langle\varphi_{jT}\rangle\Delta T+\langle\varphi_{j\eta}\rangle\Delta\eta+\langle\varphi_{jc}\rangle\Delta t+\frac{1}{3}\Delta S$$

$$(j=x,\ \Theta).$$

The integration constants of the system of equations (8) and (9) are determined from the periodicity condition

$$\Delta_n w(\Theta)=\Delta_n w(\Theta+2\pi);\ \Delta_n v(\Theta)=\Delta_n v(\Theta+2\pi) \tag{11}$$

and the conditions for the random translational motion and rotation of the fuel element as a whole:

$$\begin{aligned}\Delta_n w(\Theta=\Theta_0)&=\Delta_n w_0;\\ \Delta_n\varphi(\Theta=\Theta_0)&=\Delta_n\varphi_0.\end{aligned} \tag{12}$$

Moreover, the equilibrium condition

$$r\int_0^{2\pi}d\Theta\int_{-\delta_1}^{\delta_2}\Delta_n\sigma_x\,dz=\Delta_n N_x, \tag{13}$$

where N_α is the longitudinal force, is satisfied.

The stress increments are determined by the expressions

$$\Delta_n\sigma_x=E^*\left[\mu_x\Delta_n\varepsilon_{x0}+\Delta_n\varepsilon_{\Theta0}+\frac{z}{r}\cdot\frac{d\Delta_n\varphi}{d\Theta}\right]-\Delta_n\sigma_x^*;$$

$$\Delta_n\sigma_\Theta=E^*\left[\mu_\Theta\Delta_n\varepsilon_{\Theta0}+\Delta_n\varepsilon_{x0}+\mu_\Theta\frac{z}{r}\cdot\frac{d\Delta_n\varphi}{d\Theta}\right]-\Delta_n\sigma_\Theta^*; \tag{14}$$

$$\Delta_n\varepsilon_{\Theta0}=\frac{1}{r}\left(\frac{d\Delta_n v}{d\Theta}-\Delta_n w\right);\quad \frac{d\Delta_n\varphi}{d\Theta}=\frac{1}{r}\left(\frac{d^2\Delta_n w}{d\Theta^2}+\frac{d\Delta_n v}{d\Theta}\right);$$

$\varepsilon_{\Theta0}$ and ε_{x0} are the strains of the middle surface.

The system of equations (8), (9) is solved analytically by expanding in Fourier series certain combinations of the coefficients (10).

At each loading stage, the increments of the stress components are determined by using the stress values at the end of the preceding stage, while the parameters are determined more accurately by using the method of successive approximations [13].

Computer programs have been developed for investigating the kinetics of the stressed – strained state of fuel element jackets under transient conditions as well as under steady operating conditions of the reactor.

Calculation Results

An estimate of the efficiency of fuel element jackets based only on stresses due to a nonuniform temperature distribution along the perimeter (without an allowance for the swelling of steel) has shown that this nonuniformity plays a considerable role at relatively low temperatures and pressures. At high pressures and temperatures, the stresses due to a nonuniform temperature distribution along the perimeter relax rapidly and contribute little to the susceptibility to damage.

If the nonuniformity of the temperature distribution along the fuel element perimeter is slight (centrally located fuel elements), a swelling of steel that is almost axisymmetric hardly affects the mechanical deformation of the jacket in the case of a gaseous model and only provides a certain contribution to stresses as a result of nonuniform swelling due to the temperature drop along the thickness of the jacket.

In the case of rigid contact between the fuel and the jacket, axisymmetric swelling of steel can play a considerable role in reducing the mechanical stresses and strains in the jacket.

The peripheral nonuniformity of steel swelling can produce additional mechanical stresses and strains in both models.

We shall provide an estimate of the efficiency of jackets of the fuel elements near the walls in the monocarbide zone of the BR-5 reactor with an allowance for the nonuniform swelling of steel. Cylindrical fuel element rods with the UC fuel in the shape of baked tablets and steel 0Kh18N9T jackets are considered. A fairly large clearance, which persisted until the end of the run, was provided between the fuel and the jacket. The peak value of the temperature nonuniformity along the perimeter of a fuel element was 70°C [12], which corresponded to axial stresses of 10 kg/mm^2 in straight fuel elements. The calculations were performed for a 5% depletion, while the integral flux with respect to fast neutrons with $E > 0.1$ MeV was $\sim 3 \cdot 10^{22}$ $neutrons/cm^2$.

Figure 1 shows the variation of the operating axial stresses in the hot and the cold sides of a jacket for the median section along the fuel element, which were calculated on the basis of the above theory. The solid curves represent the stresses with an allowance for the nonuniform swelling of steel due to irradiation; the dashed curves indicate the stress variation in time without an allowance for the nonuniform swelling of steel. It is obvious from the diagram that the pattern of the jacket's stressed state changes considerably during a run as a result of nonuniform swelling.

Analysis of the efficiency of fuel element jackets shows that nonuniform swelling of steel reduces the jacket efficiency; the stress safety factor is reduced from $K_\sigma = 2.1$ (without an allowance for the swelling of steel) to 1.04. A safety factor close to unity indicates the possibility of fuel element failure. Actually, large-scale tests of the airtightness of packages [18] with fuel element jackets made of 0Kh18N9T steel (depletion ≥ 4.5%) have shown that all of them had defective fuel elements.

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INVESTIGATION OF PROMISING URANIUM CARBIDE AND PLUTONIUM CARBIDE FUEL COMPOSITIONS FOR FAST REACTORS

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In the first industrial reactors using fast neutrons the fuel used is uranium dioxide, or else a mixture of uranium dioxide and plutonium dioxide, which provides reliable fuel-element operation under conditions of high stress (high power density, fairly high temperatures, and high burnup values).

However, because of the nuclear-physics properties of oxide fuel, it is not the best fuel for the purpose.

It has been shown by calculations that the use of carbide fuel may reduce the fuel doubling time by a factor of about 1.5. Consequently, in recent years serious research has been done on carbide fuel both in the Soviet Union and abroad.

In a number of countries (France, England, the Federal Republic of Germany, the United States, and others) extensive carbide-fuel research programs are being carried on, and designs for reactors using carbide fuel have been worked out (MFR, KNK).

Operating experience with the carbide zone of the BR-5 reactor in our country has shown that there is a real possibility of using carbide fuel in fast reactors. At the same time, it has become evident that there is a need to improve the methods for obtaining monocarbides with compositions close to stoichiometric, to investigate the possibilities of suppressing the harmful effect of carbon by alloying or by applying protective coatings, and to study how the processes of gas formation and swelling vary as a function of temperature and burnup.

In comparison with oxide fuels, carbide fuels have the advantages of higher thermal conductivity and a higher percentage of the fissionable isotope. The disadvantage is that carbide fuels swell more under irradiation at temperatures above 1500°C. Therefore the advantages of carbide fuel can be utilized only if the fuel element is properly designed. In order to reduce swelling, it is desirable to use a fluid with good thermal conductivity (for example, sodium) between the core and the jacket, which considerably lowers the temperature drop and thus lowers the temperature at the center of the core. But using sodium involves a new problem: the transfer of excess carbon to the jacket and the carbidization of the jacket.

TABLE 1. Composition of Carbides Obtained by Different Methods

Method of obtaining the carbides	Chemical composition, % by weight		
	carbon	oxygen	nitrogen
Carbidization of uranium with gas	4,80—4,85	0,01—0,05	0,02
Carbidization of uranium oxides	4,80—4,90	0,01—0,1	0,05
Carbidization of plutonium solid carbon	4,2—4,6	0,01—0,04	0,04

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TABLE 2. Results of Tests on the Compatibility of Carbide Compositions with 0Kh16N15MZB Steel (Before irradiation)

Fuel	Medium	Temperature, °C	Duration of tests, h	Depth of interaction, μ	Oxygen content of the steel, % by weight		Ultimate strength, kgf/mm²		Elongation, %		Microhardness, kgf/mm²	
					initial	after tests	initial	after tests	initial	after tests	initial	after tests
UC (5.1% by weight)	Sodium	800	4500	100	0.08	0.77	65	66	45	10	210	260
UC (4.8% by weight)	"	800	2500	-	0.06	0.06	65	57	45	43	220	220
UC (5.1% by weight)	Helium	800	4000	-	0.06	0.06	65	64	45	48	210	200
UC (4.8% by weight)	"	800	2000	-	0.06	0.06	65	58	45	49	200	190
$UC_{0.8}Pu_{0.2}C$ + (5% M_2C_3 by volume)	Sodium	800	2500	10	0.07	0.14	60	60	45	35	210	230
UC (5.1% C by weight) + 5% Cr by weight	"	800	2500	-	0.07	0.10	65	67	45	36	210	220
UC (5.1% C by weight) + 9% Cr by weight	"	800	4000	-	0.07	0.08	65	65	45	45	200	200
UC (5.1% C by weight), coated with chromium	"	800	4000	-	0.07	0.12	65	66	45	42	200	210

However, there is a way to avoid the harmful effect of the carbon by working out improved techniques for obtaining carbide in stoichiometric proportions, by alloying the carbide with elements which bind the free carbon, or by applying protective coatings. Tests must be made under irradiation conditions to determine which of these methods is most effective.

Obtaining Monocarbides of Uranium and Plutonium and Mixtures of These

Specimens for the prereactor and reactor tests were prepared by the familiar methods of powder metallurgy [1, 2] – by pressing and sintering, and also by hot pressing. By these methods, it was possible to obtain cores with prescribed dimensions which had densities of 12.7-13.0 g/cm³.

Powders of the compounds were obtained by two methods: carbidization of the oxides by solid carbon, and gas carbidization of the metals.

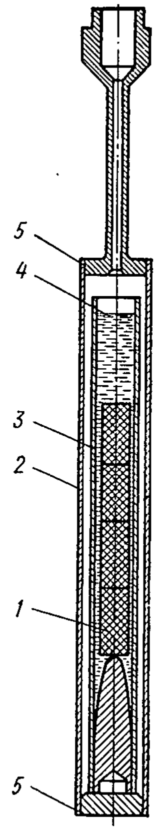
The compositions of the carbides obtained by the different methods are shown in Table 1. Figure 1 shows the typical microstructures of specimens made of uranium monocarbides and of a UC + PuC mixture. As can be seen from these data, by the use of the above-mentioned methods it is possible to obtain a carbon content which is close to stoichiometric and to keep the oxygen content between 0.01% and 0.1% by weight. These quality indicators of the resulting materials are not limiting values and may be improved through further improvements in technology.

The protective coatings were applied to the cores by electron-beam spraying in a vacuum, which made it possible to obtain uniform coatings (25-35 μ thick) that adhered well to the substrate [3].

One of the main questions involved in the possible use of carbides in industrial reactors, which have considerably higher temperatures and burnup values than the BR-5, is whether the fuel is compatible with the jacket, whether it swells, and whether it generates gas. This compatibility was investigated under both prereactor and reactor conditions.

Investigation of the Compatibility of Carbides with the Jacket Materials (0Kh16N15MZB Steel)

Investigation before Irradiation in the Reactor. The compatibility of the carbides was investigated on special specimens simulating fuel elements (Fig. 2). The compatibility investigations included microstructure, x-ray structure, and chemical analyses, as well as determinations of the mechanical properties of the jacket. The results of the tests, conducted at 800°C and lasting



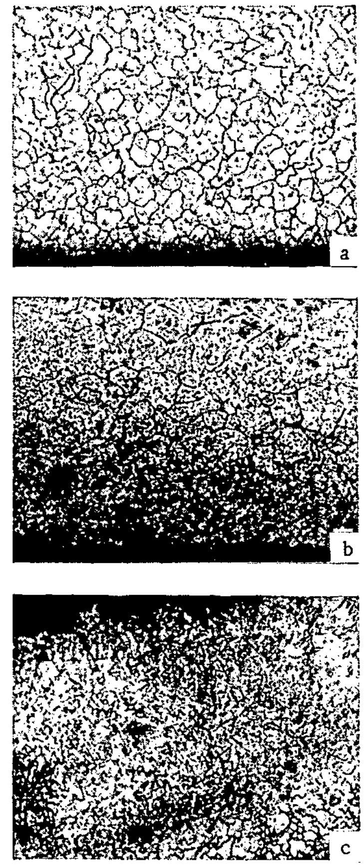


Fig. 1 Fig. 2 Fig. 3

Fig. 1. Microstructures of uranium and plutonium monocarbides (× 200): a) uranium monocarbide obtained by carbidization of uranium with propane gas (stoichiometric); b) uranium monocarbide obtained by carbidization of uranium dioxide with carbon (superstoichiometric); c) mixture of uranium and plutonium monocarbides obtained by carbidization of oxides.

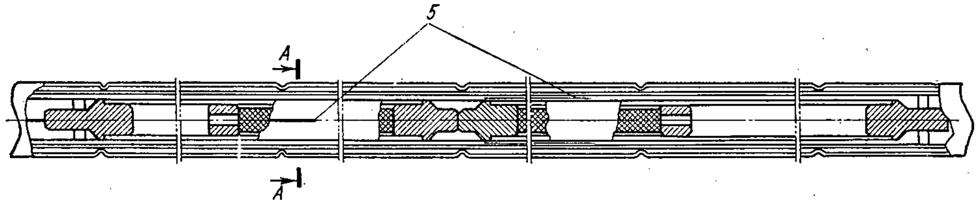
Fig. 2. Specimen for studying compatibility: 1) carbide pellets; 2) outside container made of 0Kh18N10T steel; 3) jacket made of 0Kh16N15MZB steel; 4) sodium level; 5) weld.

Fig. 3. Microstructure of a fuel element jacket (0Kh16N15MZB steel) after testing (core made of uranium monocarbide, 5.0-5.1% carbon by weight, × 200). Before irradiation: a) kept at 800°C for 4000 h; with a helium-filled gap, the material does not interact. b) after 4500 h at 800°C, with a sodium-filled gap, the interaction zone is 100 μ deep. After irradiation (c): operation at 640°C, burnup value 6.3 atomic percent.

up to 5000 h are shown in Table 2. It can be seen from the table that if the gap is filled with helium, carbide of stoichiometric and superstoichiometric composition does not interact with 0Kh16N15MZB steel. If the gap is filled with sodium, carbide of stoichiometric composition does not interact with the jacket, but carbide of superstoichiometric composition does interact, forming a layer approximately 100 μ thick; the carbon content of the steel is increased to 0.77% by weight. This is accompanied by an increase in the microhardness from 210 to 260 kgf/mm^2 and by a decrease in elongation from 45% to 10%. The interaction structure can be seen in Fig. 3.

Alloying the superstoichiometric carbide with chromium led to improved compatibility in the presence of sodium, and the increase in the chromium content from 5% to 9% by weight had a favorable effect on compatibility.

The application of a chromium coating to the superstoichiometric carbide pellets prevented the transfer of carbon through the sodium to the jacket. The mechanical properties remained at their original level.



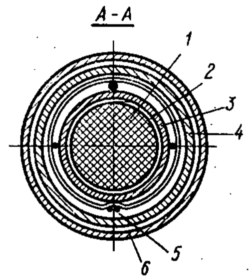


Fig. 4. Ampule for testing in the SM-2 reactor: 1) core; 2) gap; 3) jacket; 4) outside can; 5) thermocouple; 6) channel.

The compatibility of (UPu)C + 5% M_2C_3 by volume was found to be better than that of UC + UC_2. Apparently the tendency of the sesquicarbide phase to transfer its carbon is not so strong as that of the dicarbide phase.

Investigation after Irradiation in the Reactor. In order to study the behavior of the carbide fuel during irradiation, we developed a special ampule which enabled us to measure the jacket temperature during the irradiation process (Fig. 4). The fuel element is designed with a porosity of approximately 20%, consisting of the volume of the gap between the jacket and the fuel and the volume of the pores in the pellets. The gap measured 0.18-0.4 mm diametrally, and the effective density was 10.92 g/cm^3.

After irradiation, carbidization is observed in the jacket, the depth of interaction depending on the composition of the carbide, on the irradiation temperature, and on the medium. In the specimens with a sodium-filled gap the carbidization is more intense.

Table 3 shows the data on the compatibility of uranium carbides with 0Kh16N15MZB steel during irradiation. It can be seen from the data that under reactor conditions the temperature at which the interaction begins becomes 200-250°C lower than in the tests conducted prior to placement in the reactor. Thus, carbidization of the jacket to a depth of 100 μ was observed in the reactor at 640°C (see Fig. 3c), but when there was no irradiation the interaction was not observed below 800°C. It should be noted that under reactor conditions, carbidization of the jacket also takes place in the presence of helium, probably because of the transfer of carbon through the gaseous phase (CO).

A chromium coating prevents interaction both under prereactor conditions and under reactor conditions. The protective layer of chromium was satisfactorily maintained after irradiation. In addition to compatibility, after irradiation the swelling of the specimens, their density, and the volume of gases

TABLE 3. Results of In-Channel Tests on the Compatibility of Uranium Carbides with 0Kh16N15MZB Steel (After irradiation)

Fuel	medium	Temperature °C	Duration of tests, h	Depth of interaction, μ	Ultimate strength, kgf /mm²		Elongation, %		Microhardness, kgf /mm²		Remark
					initial	after testing	initial	after testing	initial	after testing	
UC (5.1%C by weight)	Sodium	500	6500	400	—	—	—	—	210	210	In the SM-2 reactor
UC (5.1% C by weight)	Helium	500	6500	100	—	—	—	—	220	290	The same
UC (5.4%C by weight)	"	570	7300—29300	10—30	73,5 *	85,0	47,0	1,3	240	400	In the BR-5 reactor
UC (5.0% C by weight)	"	640	6500	100	—	—	—	—	220	400	The same
UC (5.1% C by weight) — with chromium	Sodium	460	6500	None	—	—	—	—	220	290	

* The tests were conducted on annular specimens [4].

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TABLE 4. Conditions and Results of Irradiation in the SM-2 and BR-5 Reactors

Characteristics of specimens					Test conditions					Results of tests				
number of bundle (P) or fuel element (T)	carbon content, % by weight	fuel density, g/cm³	material of jacket	Medium in the gap	reactor	maximum linear power, W/cm	time of operation, months	temperature at center of fuel (initial), °C	temperature of jacket (calculated), °C	maximum burnup, % of heavy atom	swelling per 1% burnup	gas generation	increase in diameter of jacket	depth of interaction zone, μ
P-7	5.3-5.4	12.25-12.45	0Kh18N9T steel	Helium	BR-5	426	12	1030	570	0.8	1.2	2	0*	10
P-13	5.3-5.4	12.25-12.45	The same	The same	BR-5	426	32	1030	570	2.1	1.2	2	0-0.5†	10
P-12	5.3-5.4	12.25-12.45	" "	"	BR-5	426	47	1030	570	3.0	1.2	2	0	10
T-1	5.15	12.6	0Kh16N15 MZB steel	"	SM-2	360	9	980	500	3.5	1.2	-	0	10
T-2	4.8-5.2	12.5	The same	"	BR-5	300	-	1250	530	4.2	2.5	7	0-0.7‡	30
T-3	4.95	12.6	" "	"	SM-2	605	9	1650	640	6.3	3.5	20	1.3	100
T-4	5.1	12.5	" "	Sodium	"	360	9	1000	500	3.5	1.2	2	-	400
T-5	5.1 Chromium coating	12.6	" "	"	SM-2	254	9	880	460	3.5	-	0	-	-

* Deflections of 2 to 4 mm over a length of 500 mm were observed in all the elements of bundles P-7, P-13, and P-12, apparently because of the circular nonuniformity of the temperature field of the fuel elements near the wall.

† Deformation of jackets in the fuel element where the space between the fuel and the jacket is nearly zero was found to be 0.5%.

‡ The deformation in the jackets on individual segments of the fuel element is sometimes as high as 0.7%.

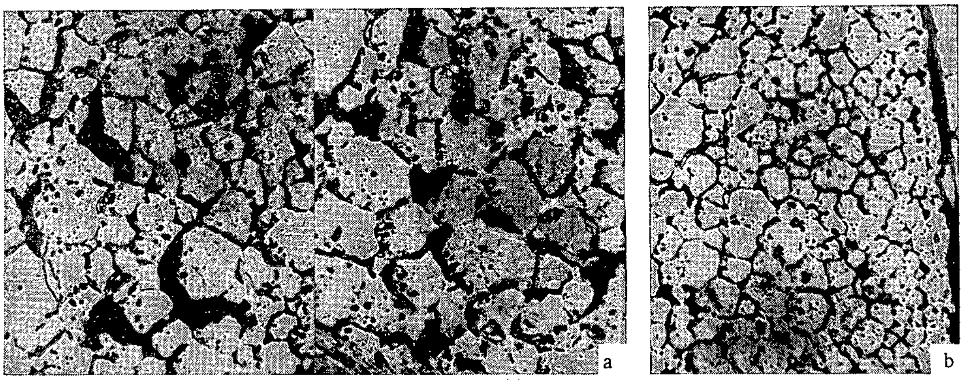

Fig. 5. Microstructure of uranium monocarbide after irradiation (burnup value 6.3 atomic percent, × 200): a) center; b) periphery.

generated were also determined. The irradiation conditions and the main results are shown in Table 4. For purposes of comparison, individual data on the irradiation of uranium carbides in the BR-5 reactor are also shown [4].

The specimens were irradiated at a linear power of 300-600 W/cm to a burnup value of 0.8-6.3 atomic percent. The maximum gas generation value was approximately 20%.

After irradiation at the conditions indicated, all the fuel elements remained hermetically sealed. The increase in jacket diameter in the fuel elements with the largest burnup values amounted to about 1%. The average rate of swelling of the carbide fuel depends very much on its temperature, amounting to 3-4% per 1 atomic percent burnup at 1400-1500°C, and the swelling is accompanied by the formation of large gas bubbles and cavities (Fig. 5). A disappearance of the dicarbide component and an increase in the number of large pores in the central part of the core have been observed.

There are few pores in the core in its initial condition, and they are distributed uniformly over the interiors and boundaries of the grains. After irradiation the pores are concentrated more along the grain boundaries. There are more large pores at the "hotter" boundaries, a fact apparently due to their migration under the influence of the temperature gradient.

Knowing the rate of swelling, we can estimate the amount of porosity required to ensure normal operation of the fuel element. By placing a sodium interlayer in the fuel element between the fuel and the jacket, we can reduce the fuel temperature, which will probably make it possible to obtain burnup values of approximately 10 atomic percent at high thermal loads. It is a more difficult problem to attain such burnup values in the fuel elements when there is a gaseous interlayer between the fuel and the jacket. Experimental data indicate that these fuel elements are capable of operating up to burnup values of approximately 6 atomic percent, but the jacket undergoes near-critical deformation. Higher burnup values can be achieved by decreasing the effective density of the fuel and also by varying the distribution of the initial porosity.

These and other questions have been included in the program of further research to be conducted on the BOR-60 reactor, and the utilization of promising types of high-temperature uranium – plutonium fuels will depend on the answers obtained.

The following conclusions may be drawn from the foregoing discussion:

1. Investigations on carbide fuels under the operating conditions of industrial reactors have demonstrated a real possibility of using such fuel at parameter values higher than those in the BR-5 reactor. No destruction of the fuel elements was observed at burnup values of up to 6.3 atomic percent and a jacket temperature of 640°C.

2. In a study of the compatibility of uranium and plutonium carbides with fuel element jackets it was found that the interaction under prereactor conditions is of the same nature as under reactor conditions. However, the temperature at which the interaction between uranium carbide and the jacket begins is 200-250°C lower under irradiation conditions. The interaction is manifested in the carbidization of the jacket to a depth of 100-400 μ, but this does not lead to any destruction of the fuel elements. The compatibility may be improved by applying protective coatings to the core.

3. The maximum rate of swelling is 3.5% per 1 atomic percent burnup when the temperature at the center of the core is higher than 1500°C, and the gas generation at a burnup value of 6.3 atomic percent is 20%.

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SOME QUESTIONS CONCERNING THE MANUFACTURE OF FUEL ELEMENTS FOR FAST REACTORS

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UDC 621.039.54

The problem of making fast reactors economical is closely related to the problem of manufacturing fuel elements capable of operating under high stress and at high fuel burnup values. What makes the problem so difficult is that such a fuel element must combine high energy generation (up to 100,000 MW · days /ton) with a considerable accumulation of secondary nuclear fuel. The fuel element must be constructed of materials which, on the one hand, have good resistance to corrosion and good mechanical properties and, on the other hand, have acceptable nuclear properties.

The first Soviet fast reactors used oxide fuels (UO_2, $UO_2 - PuO_2$) in a stainless steel jacket. This choice was made because oxide fuel has good operating characteristics and is compatible with stainless steels and sodium and because the technology of core manufacture is relatively simple, so that it is possible to design reliable and inexpensive fuel elements.

In the Soviet Union fuel elements with oxide cores have been designed for the BOR-60 and BN-350 reactors, and the manufacture of fuel elements for the BN-600 reactor is now nearing completion. These designs were preceded by careful investigations which made it possible to find the solutions that were optimal for the current state of the art. The present article explains the design approach to the manufacture of fuel elements for fast reactors, using the BOR-60 and BN-350 reactors as examples. We discuss in detail the

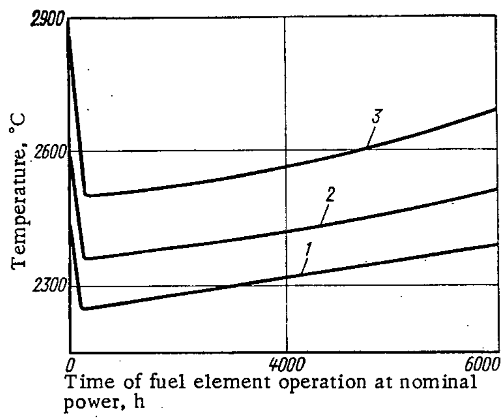

Fig. 1

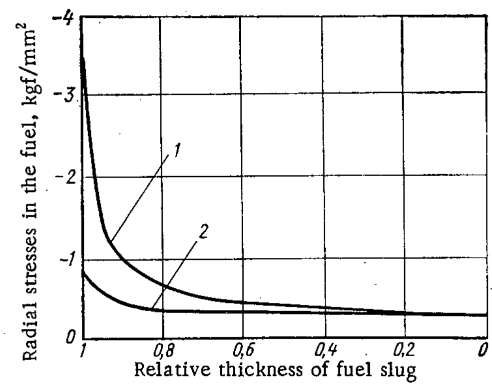

Fig. 2

Fig. 1. Change in maximum core temperature in the fuel elements of the BOR-60 reactor during a reactor run (overheating factors were taken into account in the calculations): 1) γ_{eff} = 73% of TD; 2) γ_{eff} = 80% of TD; 3) γ_{eff} = 85% of TD, W_l = 630 W/cm (TD means the theoretical density).

Fig. 2. Distribution of radial stresses at two cross sections in the fuel elements of the BOR-60 reactor: 1) cross section 0, W_l = 344 W/cm; 2) cross section 100, W_l = 482 W/cm.

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questions of choosing the effective fuel density in the fuel elements, the influence of thermal stress, pellet porosity, and jacket thickness on the deformation of the jacket due to the mechanical action of the swelling core, as well as the possible design solutions which can reduce this deformation. We consider a number of general questions related to the manufacture of fuel elements with a fuel carbide base for fast reactors. The use of UC and UC–PuC is discussed as one possible way of improving the economic indicators of fast reactors.

The Choice of Effective Fuel Density. The most important parameter of fuel elements with fuel oxides is the value of the effective density. This determines:

the amount of fertile material placed in the active zone of the fuel element, for a specified plutonium content in the fuel or for a specified degree of enrichment;

the free volume* in the fuel element that is required to compensate for the swelling of the core when fission fragments accumulate in it;

the level of operating temperatures in the core when the fuel element is in use.

The physical and economic characteristics of a fast reactor improve as the effective fuel density in the fuel element is increased, since an increase in the raw material contained in the active zone leads to an increase in the internal reproduction factor. However, in order to give the fuel element reliable operating characteristics, the effective density must be kept limited. These operating characteristics include the following:

the free volume in the fuel element must be sufficient to compensate for the volumetric changes in the core when fission fragments accumulate in it to a level of 100 kg/ton of fuel charge;

the operating temperature of the fuel, taking into account the changes in the shape of the core, must not exceed the level at which appreciable axial transfer of the fuel mass can take place;

the mechanical stresses exerted on the jacket by the swelling core must not exceed the allowable limit.

The porosity value required to compensate for the volumetric changes in the core when it swells have been estimated on the basis of experimental data obtained from the irradiation of experimental fuel elements in the SM-2 and BR-5 reactors. It has been shown that the average rate of swelling of UO_2 and UO_2-PuO_2 is approximately 1.0% for each 1 atomic percent of burnup. It follows from this that in order to obtain an energy generation value of 100,000 MW · days/ton the free volume in the fuel elements must amount to at least 12%. In choosing this volume percentage we must also take into account the tolerances in the geometric dimensions and the density of the sintered pellets, as well as the geometric dimensions of the tubes used for the fuel element jacket.

Some of the parameters of the fuel elements in the BOR-60 reactor are given below:

Outer diameter:	
core	5.2-0.2 mm
fuel element jacket	6.0 ± 0.04 mm
Diameter of axial hole in core	1.7 ± 0.12 mm
Thickness of jacket wall	0.3 ± 0.03 mm
Density of sintered pellets	91-96% of theoretical value
Calculated value of free volume in the active zone of the fuel element	26 ± 9%

It can be seen from these data that the tolerances in the tubes and the sintered pellets result in a ±9% variation in the value of the free volume in the fuel element active zone. If the effective fuel density in the fuel element has the usual value, which is 80% of the theoretical value, the free volume at individual cross sections at different heights along the fuel element may amount to 11%, which is insufficient to compensate for the volumetric changes in the core when the fragments accumulate to 100 kg/ton.

A second important requirement is to make the temperature regime in the fuel elements such that there is no axial fuel mass transfer; this requirement must be carefully considered in choosing the value of effective fuel density in the fuel element.

* The expression "free volume" means the entire free space in a cross section of the fuel element, including the gaps between the jacket and the core, the porosity of the core, and its axial hole.

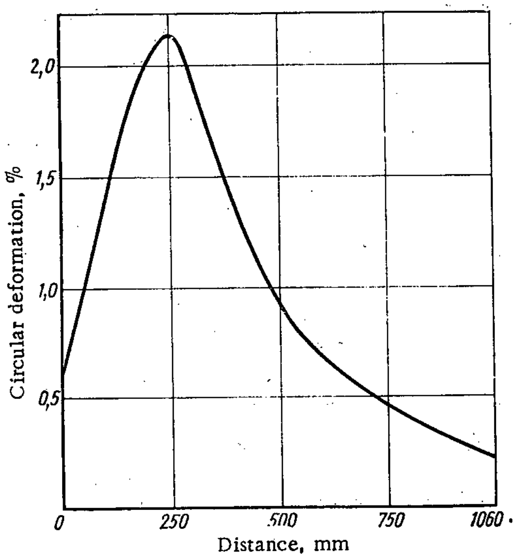

Fig. 3

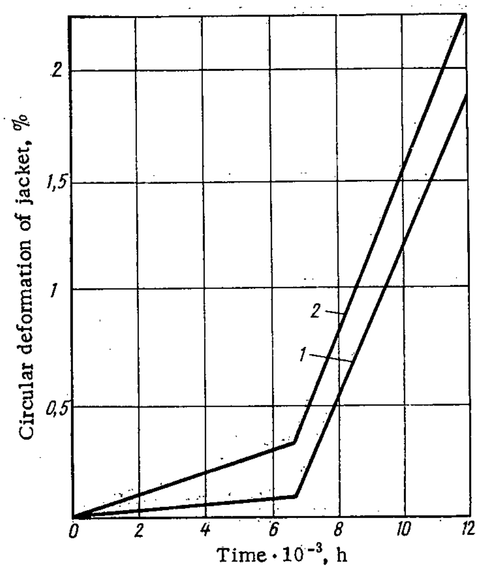

Fig. 4

Fig. 3. Deformation of the jacket as a function of distance along the fuel element in the BN-350 reactor.

Fig. 4. Deformation of fuel element jacket in the BN-350 reactor during a run (when the reactor is operated at nominal power) for a cross section 250 mm from the lower end of the active zone of a fuel element with a density of: 1) 8.25 g/cm^3; 2) 9.3 g/cm^3.

During the initial irradiation period, because of the radial mass transfer of fuel, an axial cavity is formed in the core (or an existing axial cavity becomes larger), and the diameter of this cavity can be determined from the following system of equations:

$$\rho_0 = \sqrt{\frac{\gamma_p - \gamma_{eff}}{\gamma_c} + \frac{\gamma_c - \gamma_p}{\gamma_c}\rho_c^2};$$

$$\int_{T_R}^{T_{rc}} \lambda(T)\,dT = \frac{W_l}{4\pi(1-\rho_0^2)}\left[1 - \rho_c^2 - 2\rho_0^2 \ln\frac{1}{\rho_c}\right].$$

where $\rho_0 = r_0/R$ is the relative thickness of the fuel slug; r_0 is the radius of the resulting axial cavity; R is the inner radius of the jacket; γ_p is the density of the sintered pellets; γ_{eff} is the effective fuel density in the fuel element; γ_c is the density of the fuel in the layers in which the core temperature is greater than 1700°C (on the basis of experimental data this is taken to be 98% of the theoretical density); $\rho_c = r_c/R$; r_c is the radius corresponding to a temperature of 1700°C; W_l is the specific linear power.

This system of equations is valid for a regime of fuel element operation in which there is no axial transfer of fuel mass.

During a reactor run there is a gradual change in the shape of the core as a result of fuel swelling. Depending on the ratio of jacket strength to core plasticity, as the fission fragments accumulate in the fuel, either the axial cavity will become smaller or the jacket will be deformed. If the thermal loads taken from the fuel element remain constant during the reactor run, a reduction of the axial cavity in the core will lead to an increase in the fuel temperature. These conditions of fuel element operation are found in fast reactors whose internal reproduction factor is close to unity or in reactors whose active zone contains highly enriched nuclear fuel.

The value of the effective density in these fuel elements must be so chosen that the core operating temperatures will not exceed the melting point of the fuel at any time during the operating period of the fuel element. It must be borne in mind that when a large amount of fission fragments accumulate in the fuel (up to 70-100 kg/ton), the melting point of the fuel is appreciably reduced (by 100-150°C).

Figure 1 shows the calculated results indicating the change in the maximum temperature of a fuel element core in the BOR-60 reactor during a reactor run for different values of effective fuel density.

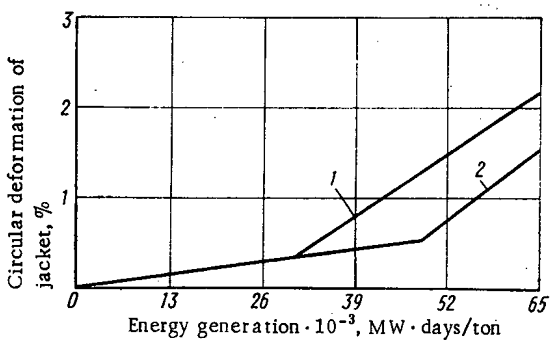

Fig. 5

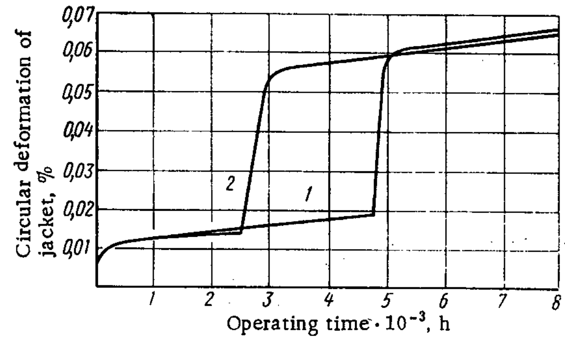

Fig. 6

Fig. 5. Deformation of the jacket of a fuel element in the BOR-60 reactor during a run for the following values of sintered-pellet density (cross section 0): 1) 10.6 g/cm^3; 2) 10.3 g/cm^3.

Fig. 6. Deformation of fuel element jacket in the BOR-60 reactor during a run for the following sintered-pellet densities (cross section 75): 1) 10.3 g/cm^3; 2) 10.6 g/cm^3.

During the initial irradiation period the temperature decreases, since the axial cavity of the core is enlarged as a result of radial fuel mass transfer. The subsequent gradual rise in temperature is caused by the reduction of this cavity as a result of volumetric changes in the core due to the confining effect of the jacket.

For an effective fuel density equal to 85% of the theoretical value, the core temperature reaches the melting point at energy generation values of approximately 70,000-80,000 MW · days/ton. Further operation of fuel elements with this effective density may lead to a transfer of the fuel to the lower part of the element. For this reason the upper limit for the effective fuel density in the fuel elements of the BOR-60 reactor was chosen to be 82.5% of the theoretical density of UO_2, and the average value does not go above 73.5% of the theoretical value.

In selecting the effective fuel density value, we must also estimate the stresses created in the fuel element jacket by the mechanical action of the swelling core. These stresses and the amount of deformation accumulated in the jacket by the end of the run depends to a large extent on the core temperature. Figure 2 shows the results of calculations for radial stresses arising in the core of a fuel element of the BOR-60 reactor by the end of the run. It can be seen from the figure that when the core swells, the stress on the jacket is exerted essentially by the outer layer of fuel. As the thermal stress increases, the fuel temperature will rise, the layer will become plastic, and the stresses in it will be reduced. The calculations were performed using the mechanical characteristics of 0Kh16N15MZB steel and uranium dioxide, which are given in [1-3]. The increase in the effective fuel density of the fuel element leads to an increase in the core temperature, and therefore in order to reduce the deformation of the jacket it is desirable to use the maximum effective fuel density in the fuel element.

Figure 3 shows the results of calculations for the deformation accumulated in a fuel element jacket in the BN-350 reactor by the end of the run. The maximum deformation is found in the lower part of the fuel element, where the plasticity of the fuel is low. The increase in effective fuel density in this part of the fuel element from 75% to 86% of the theoretical value reduces the deformation of the jacket from 2.3% to 1.8%, which increases the utility of the fuel element (Fig. 4). In the lower part of the fuel elements the burnup value does not go above 5-7 atomic percent, and the necessary free volume can be reduced to 8-10%, which makes it possible to increase the effective fuel density.

Influence of the Initial Porosity of the Sintered Core Pellets on the Utility of a Fuel Element with Oxide Fuel. Choice of Optimum Density of the Sintered Pellets. In the evaluation of the stresses* created in the jacket of a fuel element by the action of the swelling core, the influence of the porosity of the sintered pellets was also considered. The calculations used the data of [5]. It was shown above that when the core swells, the stress on the jacket is exerted essentially by the outer layer of fuel (see Fig. 2). The deformation of the jacket is determined by the rate of swelling of this layer of fuel and by its plasticity. Experimental investigations of the fuel elements tested have shown that the structure and porosity of the outer layer

* The procedure for calculating the stresses and the deformation which occur in the jacket as a result of the mechanical action of the swelling core are explained in [4].

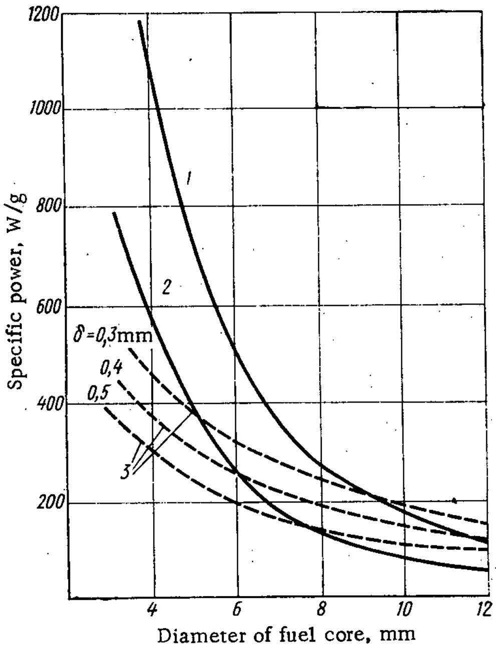

Fig. 7. Variation of the specific power obtainable from a unit weight of carbide fuel, as a function of the fuel core diameter (γ_{eff} = 80% TD; γ_p = 90% of TD): 1) sodium-filled gap; 2) helium-filled gap; 3) loads at which the stresses in the jacket are equal to the elastic limit of steel (for different values of wall thickness δ).

are not very different from the initial structure and porosity of the sintered pellets. Figure 5 shows the results of calculations of the deformation of the jacket of a fuel element in the BOR-60 reactor for different values of density of the sintered pellets. As the density of the UO_2 pellets increases from 10.3 to 10.6 g/cm^3, the deformation of the jacket in the lower part of the fuel element (cross section 0) increases from 1.5% to 2.5%. At cross section 75 (75 mm from the lower end of the active part of the fuel element) the deformations in the fuel element jackets change little with changes in pellet density (Fig. 6), since the fuel is sufficiently plastic at this cross section and relatively little stress is exerted upon the jacket by the core layers nearest the wall.

These considerations concerning the influence of porosity in the fuel layers near the wall upon the operating capacity of the fuel elements make it necessary to restrict the upper limit of the density of the sintered pellets. The upper limit chosen for the fuel density in the fuel elements manufactured for the first fast reactors was 95-96% of the theoretical value. The lower limit of the density must be selected on the basis of the condition that the fuel element must have the required fuel charge and on the basis of the technological possibilities of pellet manufacture. The minimum pellet density was chosen to be 90-91% of the theoretical value.

Comparison of Oxide Fuel Elements with Carbide Fuel Elements. In spite of the above-mentioned advantageous properties of oxide fuels, carbide fuels offer serious competition because they have the important advantages of better thermal conductivity, higher density, and fairly high melting point. When the advantages of carbide fuels are fully utilized, it is possible to improve considerably the reproduction factor and doubling time of fuel elements in fast reactors. The high thermal conductivity of the carbide fuel makes it possible to obtain higher values of specific power per unit length of core. For the same reactor power value, a smaller number of pellets is needed when carbide fuel is used, and this, to a certain extent, serves to offset the high cost of manufacturing a carbide core.

The advantages of carbide fuel are partly nullified by the fact that its radiation stability is low at temperatures above 1500°C. According to experimental data, the core temperature in fuel elements using carbide fuel must not exceed 1500°C. Above this temperature the rate of swelling of the carbide increases sharply, reaching values of approximately 3-4% volume increase for 1 atomic percent of burnup. At such a rate of swelling, fuel elements must contain 30-40% free volume in order to reach energy generation values of 80,000-100,000 MW · days/ton, and this nullifies the advantages of the carbide fuel.

However, even with such a restriction on the operating temperatures in the core, carbide fuel makes it possible to extract a substantially higher level of power from a unit length of fuel element than in the case of oxide fuel. In these cases it is more advantageous to use a sodium interlayer between the core and the jacket in carbide fuel elements. Calculations show that for the same reactor power value, the use of carbide fuel elements with a sodium-filled gap requires the manufacture of approximately one-third as many cores as in the case of oxide-fuel cores and approximately two-thirds as many in the case of carbide fuel elements with a helium-filled gap. However, as can be seen from Fig. 7, the above-mentioned advantages of carbide fuel can be attained only if the diameter of the fuel elements is sufficiently large, i.e., to the detriment of the specific power obtainable from a unit weight of fuel, which is another important factor that significantly influences the economic indicators of the entire fuel cycle. A reasonable compromise must therefore be established. Fuel elements with a helium-filled gap afford less of an opportunity to utilize the advantage of good thermal conductivity in carbide fuel. However, fuel elements of this kind are simpler to manufacture.

During the first stage of fast reactor development, when the decisive factor is the specific load drawn from a unit weight of fuel, fuel elements with a helium-filled gap will apparently be more widely used. For fuel elements with sodium interlayers it is desirable to use jackets with wall thicknesses of 0.25-0.3 mm. It is possible to relieve the jacket of the mechanical stress exerted by the swelling core if the necessary gap is left between the jacket and the fuel. The pressure of fission-fragment gases inside the fuel element can be reduced to an acceptable value either by building the necessary free volume into the fuel element or by using a fuel element design which enables the gaseous fission products to escape into the coolant loop.

The above considerations concerning the possibilities of using carbide fuel were borne in mind by the authors in the design of fuel elements with carbide cores for experimental bundles to be used in the BOR-60 reactor. These bundles, which are now being installed in the reactor for testing, contain chiefly fuel elements with helium interlayers. The cores are 5.8 mm in diameter, i.e., the dimension chosen was the one making it possible to extract the maximum specific power from a unit weight of fuel (see Fig. 7). Fuel elements with sodium interlayers and with jackets 0.25-0.3 mm thick are being designed for a second series of experimental bundles for the BOR-60 reactor.

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POST-PILE STUDIES OF FUEL ELEMENTS IN SERVICE FOR 626 EFFECTIVE DAYS IN THE CORE OF THE VVÉR-1 REACTOR OF THE NOVO-VORONEZH NUCLEAR POWER STATION

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V. Ya. Gabeskiriya, and V. F. Portnov

UDC 621.039.548

Low operating costs of nuclear power stations are determined to a significant extent by the reliability of fuel elements in service at high burnup levels. Some of the crucial factors governing the viability of fuel elements in the core are the corrosion resistance of the cladding material, oxygenation and hydrogenation of the zirconium alloy, and the distribution pattern of the hydride phase throughout the cladding.

This article reports results of an investigation of the official reactor fuel elements after service for 626 effective days in the VVÉR-1 reactor under the most severe irradiation conditions, with a peak burnup of 24,000 MW · days/ton U attained.

Design of Fuel Element and Operating Conditions. The fuel element in service in the water-cooled water-moderated reactor of the first power unit at the Novo-Voronezh nuclear power station [1] is a cylindrical zirconium (Zr + 1% Nb) tube 10.2 mm in diameter, 0.65 mm wall thickness, filled with 2% enriched sintered uranium dioxide pellets. The diametral clearance between the pellets and the tube wall is 0.1 mm. The bulk density of the sintered uranium dioxide is ~10.1 g/cm^3. The fuel element is hermetically sealed by means of stepped blind flanges made of zirconium alloy welded in place by electron-beam welding or argon-arc welding. The top of the fuel element features a free space for compensating temperature expansions of the column of pellets, and for compensating pressure built up by gaseous fission fragments.

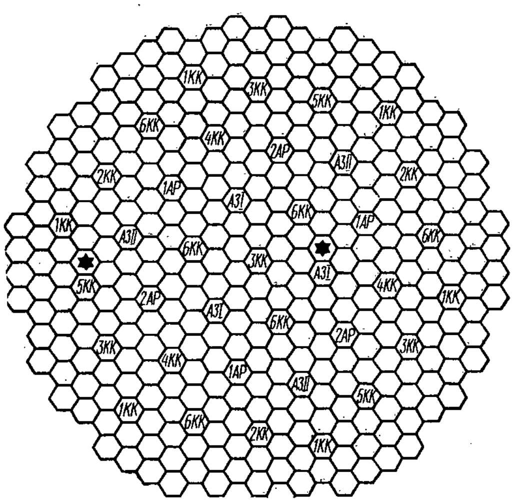

Fig. 1

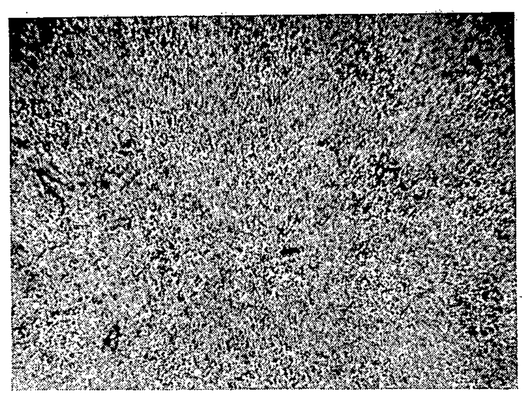

Fig. 2

Fig. 1. Arrangement of fuel elements investigated in core of VVER-1 reactor: the fuel assemblies marked by the star were investigated; A3, AP, and KK are compensating assemblies.

Fig. 2. Microstructure of cladding in original state (× 200).

Translated from Atomnaya Énergiya, Vol. 30, No. 2, pp. 222-226, February, 1971.

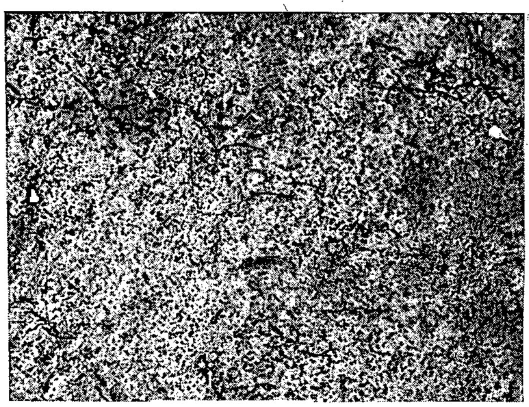

Fig. 3

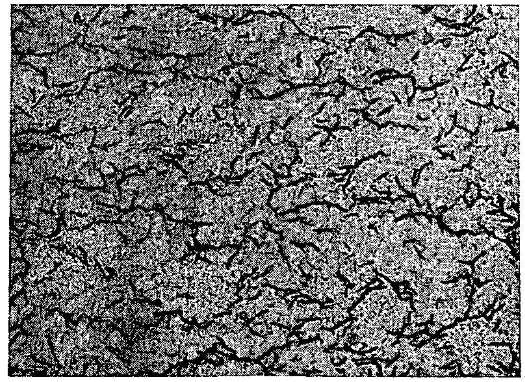

Fig. 4

Fig. 3. Microstructure of cladding, coupon cut from top of fuel element (× 200).

Fig. 4. Microstructure of cladding, coupon cut from middle of fuel element (× 200).

Each working fuel assembly contains 90 fuel elements.

Below, we cite results of an investigation of fuel elements from two assemblies in service under the most severe burnup conditions in the VVÉR-1 reactor core, from August, 1964 all the way through June, 1967. The arrangement of the fuel assemblies investigated, in the reactor core, is shown in Fig. 1. The fuel elements were cooled by water pressurized at ~100 atm. The water temperature at the entry to the core was 240-250°C, and 275-280°C at the reactor exit. The chemical composition of the water was [2, 3]: hardness ≤0.003 mg-eq/liter; oxygen content ≤ 0.015 mg/liter; chlorides ≤ 0.05 mg/liter; pH 9-10; corrosion products 1.5 mg/liter.

In the initial service period of the nuclear power station, the high pH of 9 to 10 was maintained by introducing hydrazine hydrate and ammonia into the makeup water. But it was found that large amounts of hydrogen accumulated as a result of breakdown of the ammonia and hydrazine acted upon by radiation in the core, and this buildup of hydrogen engendered a danger of formation of an explosive mixture in gas holders and in other vessels. Subsequently, the pH of the primary-loop water was allowed to drop to 6-7. The average heat loading on the surface of the fuel elements was $3 \cdot 10^6$ to $4 \cdot 10^6$ kcal/m$^2 \cdot$h.

External Inspection of Fuel Elements. Inspection of the surface of the cladding on the fuel elements was carried out in a "hot" cave, using the UMSD-1 remote-control microscope. Dark gray patches of oxide film showed up on the surface of the cladding, with no flaws of any kind (blowholes, blisters, cracks, etc.) in evidence, and separate traces of deposits of brown coloration were also observed, and attributed to presumed precipitation of corrosion products from the reactor loop (iron oxides, chromium oxides, etc.). All of the welded joints made by electron-beam welding remained in an excellent state. The outer diameters

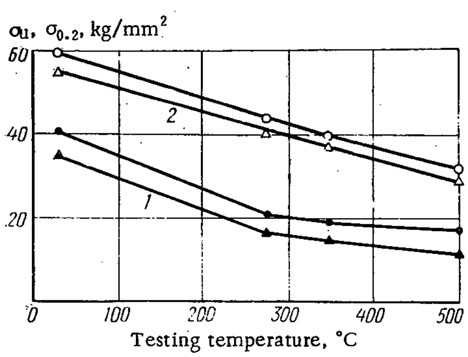

Fig. 5

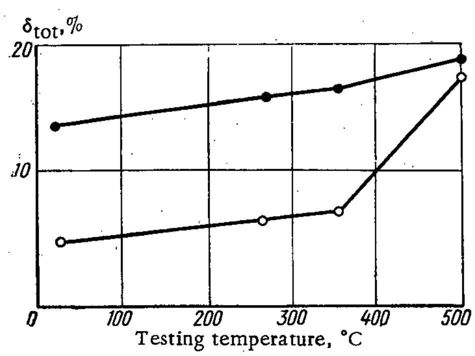

Fig. 6

Fig. 5. Dependence of strength properties on testing temperature: 1) control coupons; 2) irradiated coupons; ●○) σ_u; ▲△) $\sigma_{0.2}$.

Fig. 6. Dependence of ductility on testing temperature: ●) δ_{tot}, control coupons; ○) δ_{tot}, irradiated coupons.

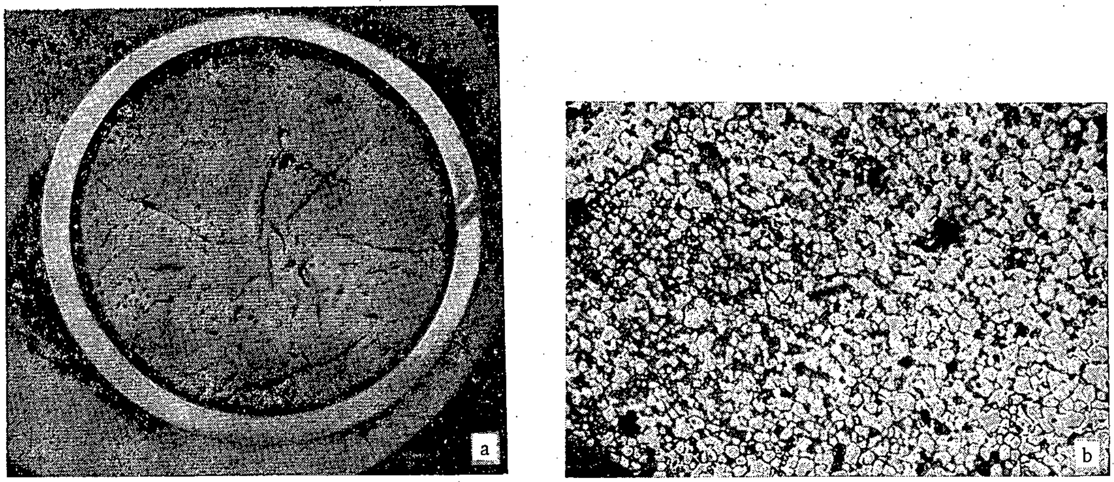

Fig. 7. Macrostructure and microstructure of the fuel (portion of fuel element subjected to highest irradiation intensity).

of the fuel elements were measured. Measurements were taken every 200 mm, in two mutually perpendicular directions, and with precision to within ±0.01 mm. These measurements revealed that the dimensions of the element remained within the original tolerance range diametrally.

Structural Investigations and Mechanical Tests. Fuel elements from each fuel assembly in outer and inner rows of the assembly, with both minimum and maximum fuel burnup, were subjected to metallographic investigation. Coupons were cut from segments spaced 400, 1000, 1300, 1700, and 2400 mm from the bottom end of the fuel elements, in order to study the microstructure of the cladding material (alloy Zr + 1% Nb) and of the nuclear fuel (UO_2). The height of the coupons was made 7 mm for the metallographic investigations, and 3 mm for the mechanical tests under tensile load. The microstructure of the cladding is shown in its original state in Fig. 2.

Metallographic investigations of the fuel-element cladding revealed the presence of a moderate amount of hydride phase in the structure. Figures 3 and 4 show typical photographs of the microstructures of coupons cut from the cladding of fuel elements at the top and in the middle, corresponding to minimum burnup and maximum burnup of the fuel respectively. It was found that an oxide film from 0.01 mm to 0.03 mm thick formed on the surface of all of the fuel-element jackets. The thickness of this oxide film increased toward the center of the fuel element. The quantity of hydride inclusions increases in the zone of maximum power release (~1000 mm from the bottom of the fuel element). The hydrides present in the cladding display an annular orientation with increasing concentration toward the outer surface.

The moderate amount of hydride phase present in the cladding, and the insignificant thickness of the surface oxide films, provide evidence of the conventional process of oxidation and hydrogenation of the zirconium jackets from the exterior, in the coolant medium, at work. The microhardness of the jackets on these irradiated fuel elements increased 60-70% over the microhardness of the fuel-element cladding jackets in the original state. A series of tests was run on unirradiated (controls) and irradiated annular coupons cut from different portions of the fuel elements, in order to obtain a quantitative estimate of the mechanical properties of the fuel-element cladding.

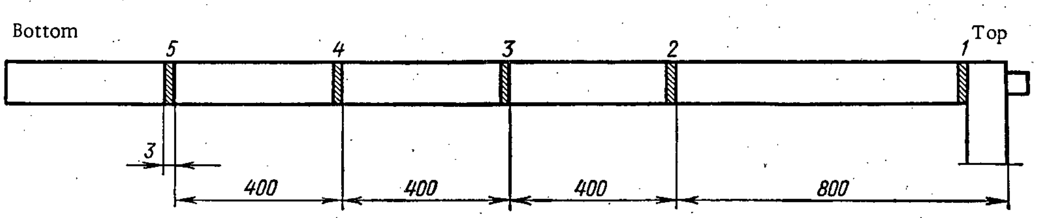

Fig. 8. Diagram illustrating where coupons were cut from fuel element.

TABLE 1. Content of Uranium and Plutonium Isotopes in the Specimens Analyzed

Number of specimen	U^{235} content, %	Content of Pu isotopes, %			
		Pu^{239}	Pu^{240}	Pu^{241}	Pu^{242}
1	1,4±0,06	82,6±0,8	13,4±0,7	3,5±0,1	0,54±0,05
2	0,78±0,02	62,2±0,4	24,05±0,16	10,14±0,25	3,6±0,11
3	0,63±0,01	60,0±0,4	25,0±0,2	10,9±0,3	4,35±0,2
4	0,76±0,03	63,6±0,6	22,6±0,4	11,2±0,6	2,54±0,04
5	0,66±0,014	60,9±0,5	24,2±0,3	10,8±0,2	4,02±0,12

Tests were conducted at temperatures 280°C, 350°C, and 500°C in vacuum (10^{-4} mm Hg), and at room temperature, in air on a remote-controlled testing machine (MM-150D) with automatic recording of the tensile loading chart. The temperature at which the tests were conducted was maintained to within ±3°C automatically.

It is clear from the data obtained from tensile loading tests on the annular coupons (Figs. 5 and 6) that the ultimate strength σ_u and the yield point $\sigma_{0.2}$ rose appreciably above the values measured in the case of jackets on unirradiated fuel elements. As a result of the irradiation, the ductility of the coupons cut from different portions of the cladding of the irradiated elements decreased by 50-60%, and amounted to 6-7% at testing temperatures from 280°C to 350°C, in the case of annular coupons.

Results of the mechanical tests, along with the data from investigations of irradiated fuel elements, attest to the excellent stability of jackets made from zirconium alloy Zr + 1% Nb.

Investigation of the Fuel. As a result of investigation of the microstructure of UO_2 taken from different portions along the height of fuel elements from fuel assemblies Nos. 1 and 2, including portions from the regions of maximum heat release, it was established that no substantial changes had taken place (see Fig. 7).

Since there are no distinct temperature zones (meltdown or fusion zone and columnar-grain zone) in the structure of the fuel meat, and since no growth of equiaxial grains is observed, the temperature at the center of the pellets could not exceed 1600°C.

Radiochemical investigations of UO_2 specimens taken from fuel elements of the fuel assemblies investigated were carried out in order to make an experimental determination of the isotope composition and burnup of irradiated fuel in the VVER-1 reactor.

Radiochemical Investigation of the Fuel. For the radiochemical investigation of the fuel, five coupons from different portions along the height of the fuel element (Fig. 8), mostly from the maximum burnup zone, were taken in order to provide information on changes in the isotope composition of the fuel at the highest burnup levels.

Method of Investigation. Determination of the isotope composition of the uranium and plutonium was carried out on MI-1311 and MI-1305 mass spectrometers [4]. The content of plutonium isotopes per gram of uranium was determined by two methods, by a coulometric technique and by the isotope dilution technique [5].

The value of the plutonium content in the solution analyzed was found as the average of two measurements obtained coulometrically and by the isotope dilution method, since the errors in these two methods are roughly the same.

Experimental Results. The experimental results of the determination of the amount of uranium and plutonium present in specimen Nos. 1-5 are listed in Table 1.

It is clear from Table 1 that the content of heavy plutonium isotopes in specimen 3 is 40%. The error in the determination of plutonium content in the specimens is ~10%.

TABLE 2. Content of Uranium and Plutonium (kg/ton U) in Specimens with Different Percentage Burnup

Isotope	Number of specimen				
	1	2	3	4	5
U^{235}	13,8	7,5	6,1	7,4	6,5
U^{238}	974,4	964,7	959,8	967,8	963,8
Pu^{239}	3,1	4,2	5,0	3,6	4,2
Pu^{240}	0,51	1,6	2,1	1,3	1,7
Pu^{241}	0,13	0,7	0,91	0,64	0,75
Pu^{242}	0,02	0,24	0,36	0,14	0,28
Products of plutonium fission	1,8	8,7	11,9	6,5	9,3
Products of uranium fission	5,2	10,4	11,7	10,6	11,3
Total burnup	7,0	19,1	23,6	17,1	20,6

Relationship between the Content of Isotopes of U and Pu and the Percentage Burnup. The quantity of fission products formed through fission of the isotopes U^{235}, Pu^{239}, and Pu^{241} (see Table 2) was determined by calculations [6], and on the basis of the experimental results detailed in Table 1.

It is clear from Table 2 that the contribution made by the plutonium to total burnup is quite substantial, attaining a level of ~50% in the coupon with the highest burnup. The total burnup in this specimen amounts to 23.6 kg/ton U, which is in satisfactory agreement with results reported earlier [7].

Investigations carried out on fuel elements from two fuel assemblies that saw service under the conditions of highest heat release rate in the core of the VVÉR-1 reactor, over a period of 626 effective days, until a burnup of 24,000 MW · days/ton U had been achieved, provide evidence of the high performance capabilities and the reliability of rod type fuel elements with cores of sintered uranium dioxide clad with zirconium alloy plus 1% niobium.

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DESIGN AND BASIC CHARACTERISTICS OF THE FUEL ELEMENT FOR THE VVÉR-1000 REACTOR

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UDC 621.039.546

The VVÉR-1000 reactor is a further development of the pressure vessel type water-cooled water-moderated reactors developed in the USSR. Table 1 gives the basic characteristics of the cores for the reactor types VVÉR-210, VVÉR-440, and VVÉR-1000 [1]. The comparison data are for the planned American reactor "Maine Yankee" to be put into operation in 1972 [2].

Design of Fuel Element and Basic Parameters

The design of the fuel element is shown in Figs. 1 and 2. The cladding material and the material for the end parts is zirconium alloy. The fuel consists of pellets of sintered uranium dioxide (bulk density not less than 10.2 g/cm^3, enrichment 2-4%) with holes on the end faces. The column of fuel weighs 1.53 to 1.62 kg. The diametral clearance between the fuel and cladding ranges from 0.14 to 0.27 mm. As the diagrams clearly show, the fuel meat is held between two split inserts capable of confining three times or four times the weight of the column, with a guaranteed allowance of 0.05 to 0.08 mm between the insert and the inner surface of the cladding jacket. The inserts are made of zirconium alloy and their function is to space the fuel during manufacturing processes and when the fuel is being shipped. The introduction of the bottom insert into the fuel-element design makes it possible to establish two degrees of freedom for thermal expansion of the fuel as the reactor is being brought up to power, and at the same time makes it possible to free the cladding from the tensile loads brought about by friction and by local wedging of the fuel. The design provides for two gas collectors (one on top, one below). The total free volume under the jacket is sufficient to allow the pressure generated by gaseous fission products released under the fuel-element jacket to remain below the level of coolant pressure throughout the reactor campaign. The basic parameters of the VVÉR-1000 fuel elements are cited below.

All the parameters are stated for the maximum load on the fuel element (at the maximum diametral clearance).

TABLE 1. Basic Characteristics of Cores of the Reactors VVÉR-1000, VVÉR-210, VVÉR-440, and the "Maine Yankee" Reactor

Parameter	VVÉR-1000	VVÉR-210	VVÉR-440	"Maine Yankee"
Power output, MW: electrical	1 000	210	440	830
thermal	3 200	760	1 370	2 440
Pressure in primary loop, kg/cm^2	160	100	125	158
Temperature of water at reactor inlet, °C	290	252	270	288
Average temperature of water at reactor exit, °C	340	272	300	317
Peak thermal flux, kcal /m^2·h	$1{,}35 \cdot 10^6$	$1{,}2 \cdot 10^6$	$1{,}0 \cdot 10^6$	$1{,}1 \cdot 10^6$*
Core-average burnup, MW·days/ton U	40 000	15 000	27 300	30 000

* Value obtained experimentally according to data in [2].

Translated from Atomnaya Énergiya, Vol. 30, No. 2, pp. 226-228, February, 1971.

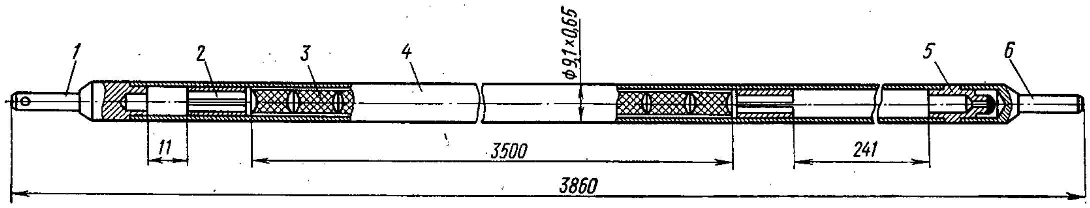

Fig. 1. Fuel element of VVÉR-1000 reactor: 1) bottom cap; 2) split insert; 3) pellet; 4) cladding jackets; 5) insert; 6) and plug.

Maximum temperature of coolant – cladding interface	350°C
Maximum temperature of inner surface of cladding (ignoring effect of corrosion film and corrosion products)	410°C
Maximum temperature of fuel surface	1270°C
Maximum temperature of fuel (at center)	2700°C
Temperature on edge of pellet, averaged over core height	1380°C
Volume-average fuel temperature	1570°C
Maximum increase in length of fuel column by end of reactor campaign	73 mm
Maximum increase in volume of fuel campaign by end of reactor campaign	10.3 cm^3
Maximum amount of gaseous fission products liberated under fuel-element jacket	54%
Free volume under jacket, in cold state	30.5 cm^3
Free volume under jacket, during service	21.8 cm^3
Maximum pressure of mixture of gases under jacket, by end of reactor campaign	140 kg/cm^2

The values of the thermal conductivity of the cladding – fuel interface and of the UO_2 used in the calculations yields fuel temperature values a bit too high, but providing the necessary safety margin in the design of the fuel element. Calculations of the volume of the gas collectors, and the diametral and axial clearances, are made on the basis of assumed uniform thermal expansion and radiation-induced swelling of the fuel exposed to irradiation. The following sequence of calculations was adopted by way of estimating the amount of gaseous fission products liberated under the fuel-element jacket: 100% of the gaseous fission products formed from the volume of fuel at temperatures above 1650°C is liberated, and 5% of the gaseous fission products from the volume of fuel at temperatures below 1650°C is liberated.

Fuel Element Fabrication Technology

The fuel-element meat is made by compacting pellets from pressed UO_2 powder. After drying, the pressed pellets are sintered, inspected externally, checked for density and chemical composition, and are then polished and inspected on their outer diameter and for traces of spalling. Serviceable pellets are loaded into the fuel-element. The assembled column of fuel is then loaded into the cladding jacket with the bottom cap already welded in place. In order to provide hermetic sealing of the fuel element, an insert with an opening through which the fuel element is filled with a mixture of inert gases (argon and helium) is welded to top end of the fuel element jacket. This opening is then welded shut, the fuel element is inspected for leaktightness at room temperature, and the end plug is welded to the insert. The end hole in the insert is sealed by argon-arc welding, and all the remaining weld seams are made by electron-beam welding. The welded joints are all annealed in order to improve their resistance to corrosive attack, after which the fuel elements go to an etching step and autoclaving step.

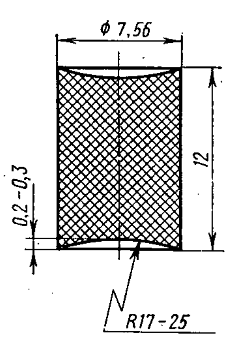

Fig. 2. Sintered uranium dioxide pellet.

The design of the fuel element for the VVÉR-1000 reactor, as discussed here, stands as the basis for further development and testing.

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INVESTIGATION OF FUEL ELEMENTS IN A FUEL ASSEMBLY IN SERVICE FOR 17,000 HOURS IN THE CORE OF THE VK-50 REACTOR

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UDC 621.039.548

An experimental boiling-water reactor, VK-50 type, has been in operation at the V. I. Lenin Scientific-Research Institute for Atomic Reactors [NIIAR], testing fuel elements clad with Zr + 1% Nb alloy. Reports on the VVÉR-68 reactor presented at the COMECON conference detailed results of investigations of fuel elements in a fuel assembly that had been in irradiation service in the reactor core for 5000 h [1]. The present report contains data on investigations of a fuel assembly that was in service in the VK-50 boiling-water reactor for a far more protracted period (17,000 h).

Operating Conditions of the Fuel Assembly

As reported in earlier articles [1, 2], the fuel assembly used in the VK-50 reactor comprises a bundle of 126 fuel pins with cores of sintered 2%-enriched uranium dioxide. The cladding of these fuel elements in Zr + 1% Nb alloy, and the jacket on the fuel assembly is made of Zr + 2.5% Nb alloy. The fuel assembly in question was loaded into the reactor core in October, 1965, and withdrawn in October, 1968. During the entire time, the operating conditions and irradiation conditions of the fuel assembly, and its position in the core, remained unaltered. Figure 1 shows the position of the fuel assembly in the reactor core.

The service time of the fuel assembly, with the reactor at different power output levels, was 17,375 h. The average integrated neutron flux was $3.8 \cdot 10^{20}$ neutrons/cm^2 (E > 0.1 MeV).

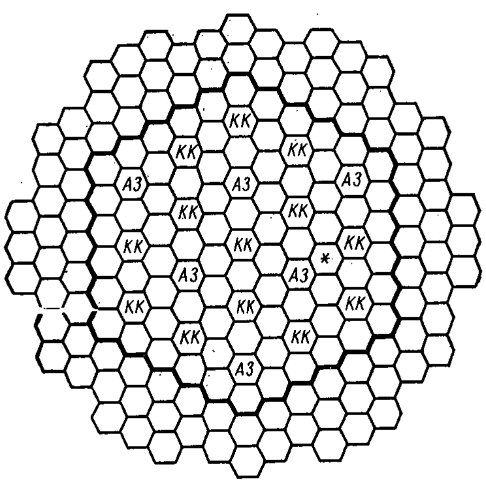

Fig. 1. Position of fuel assembly investigated in core of VK-50 reactor (marked by star).

The maximum fuel burnup was $13.4 \cdot 10^3$ MW · days/ton U. The peak thermal flux was $0.85 \cdot 10^6$ kcal/m^2 · h. The chemical composition of the boiler water (for the water conditions stabilized in steady state during the more recent years of service) is listed below [3]:

pH	6-8
Fe	0.04-0.07 mg/kg
Cu	up to 0.01 mg/kg
Zn	up to 0.015 mg/kg
O_2	up to 0.1 mg/kg
Hardness salts	10-12 mg-eq/kg

Results of the Investigation

Inspection of the fuel assembly and of the fuel elements, carried out in the primary investigations of the fuel assembly in the hot cave of the VK-50 reactor installation, and in subsequent investigations in the materials science branch, revealed traces of somewhat advanced oxidation on the jacket

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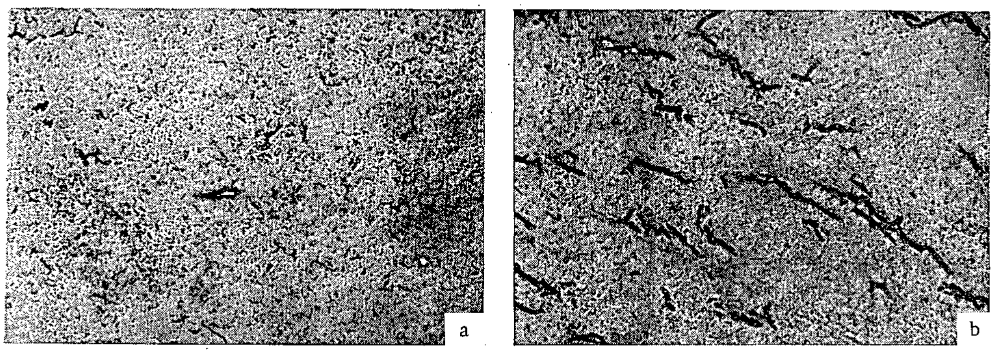

Fig. 2. Microstructure of the fuel-element cladding: a) top of fuel element; b) region of maximum heat release.

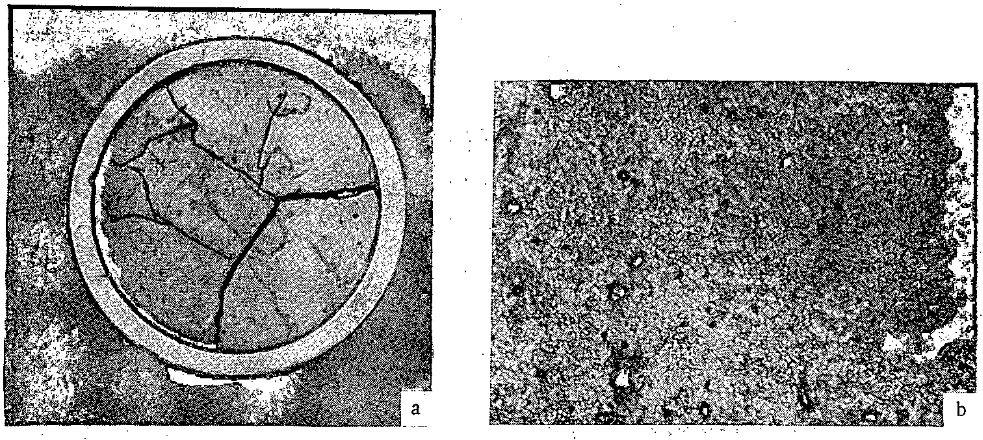

Fig. 3. Structure of fuel: a) large section through fuel meat; b) microstructure of fuel meat.

of the fuel assembly: the surface became gray, with small deposits of corrosion products from the loop. No flaws were detected on the fuel elemtns, with the exception of one element. The surface of all the fuel elements was covered by a brown incrustation of corrosion products. The thickness of these deposits varied with the height of the fuel elements, and was greatest (~0.15 mm) at those portions of the fuel elements which were in the highest irradiation and heat release regions.

Coupons were cut from the regions of maximum and minimum power release in order to facilitate mechanical and metallographic investigations of the jackets and cladding.

Photographs of the microstructure of the cladding of fuel elements in the outer row, in coupons taken from the regions of maximum and minimum (50 to 60 mm from the top of the fuel element) power release, are shown in Fig. 2.

Hydride inclusions displaying a favorable annular orientation are detected in the structure of the cladding, and appear much bigger in the region of maximum power release. The position of the hydrides in the cladding of fuel elements from the inner rows reveals the same pattern, but their content is slightly lower. The microhardness of the cladding material in the region of maximum power release is the same ~287 kg /mm^2, for both outer and inner fuel elements.

Welded seams of pressuretight parts of the fuel elements are in satisfactory condition, and show no flaws.

No drastic changes were detected in the structure of the fuel meat. The usual cracking of uranium dioxide pellets shows up in photographs of the microstructure (Fig. 3).

The mechanical tests were carried out on annular specimens ϕ 10.2 × 0.65 × 3 mm on an MM-150D remote-controlled testing machine. The testing temperatures were 20°, 280°, 350°, and 500°C. Control

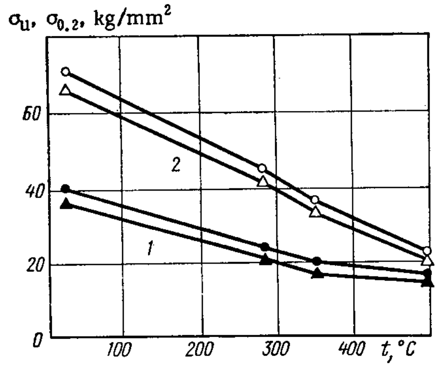

Fig. 4

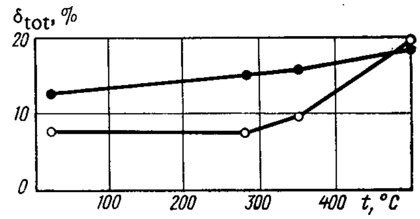

Fig. 5

Fig. 4. Dependence of strength properties on testing temperature: 1) control coupons; 2) irradiated coupons; ●○) δ_u; ▲△) $\delta_{0.2}$.

Fig. 5. Dependence of ductility on testing temperature: ●) δ_{tot}, control coupons; ○) δ_{tot}, irradiated coupons.

Fig. 6. Microstructure of portion of fuel-element cladding in the flawed region.

specimens, coupons cut from the cladding of an unirradiated fuel element, were tested at the same time. Results of the tests are plotted graphically in Figs. 4 and 5. The ultimate strength and yield point rose by 80% and 90% respectively, and the ductility at the testing temperatures of 280° and 350°C, corresponding to the service temperature, fell by 60-70%.

As noted earlier, one flawed fuel element was detected in the primary investigations. Investigation of the structure of the cladding in the flawed fuel element revealed some characteristic details:

1. Oxide film is inspected both on the outer and inner surfaces (the film thickness is 0.4 mm).

2. The cladding becomes demonstrably thinned and hydrogenated on the inner surface (Fig. 6).

3. The cladding shows "chronic" cracks with oxide film on the surface (Fig. 7).

4. The large amount of hydrides with preferential radial orientation (see Fig. 6) in the region of maximum power release, and the absence of radial orientation in the region of minimum power release.

The mechanical tests carried out on coupons cut from the failed zone showed an abrupt loss of ductility compared to the ductility of the cladding in unirradiated fuel elements. Some of the coupons experienced brittle fracture during the tests. The ductility of the cladding material at the top end of the fuel element did not change so drastically, and was similar to the ductility exhibited by unirradiated unflawed fuel elements.

Discussion of the Results

It had been reported earlier [1] that, after 5000 h of service of the fuel elements in the core of the VK-50 reactor, appreciable hardening of the zirconium alloy (Zr + 1% Nb) occurred (70-80% hardening

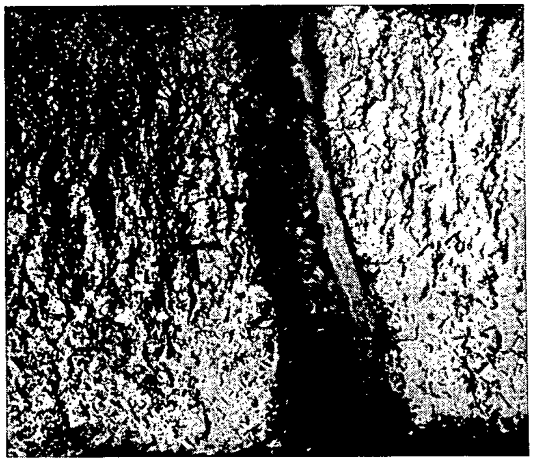

Fig. 7. Microstructure of the cladding of a flawed fuel element in the neighborhood of a crack.

was reported), with a 60-70% reduction in ductility when coupons were tested at temperatures in the range 280-350°C. Results of mechanical tests on annular coupons cut from the cladding of fuel elements that had been in service for 17,000 h in the core of the VK-50 reactor provide evidence of the same order of magnitude of hardening in the Zr + 1% Nb alloy. The ductility was retained at a fairly high level (6-7%).

On the basis of these data, we infer that, after 5000 h in service, the mechanical properties of the alloy come closer to steady-state value, while, at more prolonged service times (17,000 h) there are still no abrupt changes observed in the mechanical properties.

Comparison of results on the service life of the cladding and fuel in fuel elements in service for 5000 h and 17,000 h revealed that the pin type fuel elements of Zr + 1% Nb alloy give excellent performance in long-term service under the conditions prevailing in the VK-50 boiling-water reactor.

Metallographic data obtained in investigations of the cladding of the flawed fuel element indicate that failure of one of the 126 fuel elements occurred at the beginning of its service, as a result of pressurization failure and penetration of coolant under the cladding jacket through some kind of primary flaw, and not as a result of oxidation of the alloy by the coolant.

The investigations carried out on the cladding material and fuel material in the fuel elements used in the VK-50 reactor show that, after protracted service in the core of the VK-50 boiling-water reactor, the pin type fuel elements retain their capabilities, while cladding of fuel elements made of Zr + 1% Nb alloy are in a satisfactory condition after that time in service.

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MECHANISM OF RADIATION DAMAGE AND THE RESOURCE OF STRUCTURAL GRAPHITES AT HIGH TEMPERATURES IN LARGE NEUTRON FLUXES

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Radiation damage in graphite has been investigated over almost two decades (the first publications appeared in 1955-1956 [1-4]). Work on problems of practical importance for the reactor technology as well as general problems connected with the phenomenon from the point of view of solid state physics has been in progress.

A considerable amount of experimental and theoretical data is available at present. However, many problems of radiation damage in graphite have not been solved, which renders the prediction of results difficult. Besides the particular problems concerning changes in certain graphite characteristics as a result of irradiation, there is the unsolved problem of flaws in graphite, which determine the varying nature of damage in different temperature ranges.

This situation is due to the general difficulties in the theory of solids and also the particular difficulty of describing the structure of artificial polycrystalline graphite, which, in spite of the clearly defined structure of an ideal crystal, must be considered as a complex multiphase system.

In fact, the phenomena occurring in graphite structures as a result of irradiation at different temperatures should be considered on the basis of phase equilibrium. It would then be possible to arrive at theoretical conclusions concerning specific types of flaws and their interrelationships as well as practical technological indications on the operating resource of graphite in reactors and ways of producing graphite with the required characteristics. We shall now attempt such an analysis.

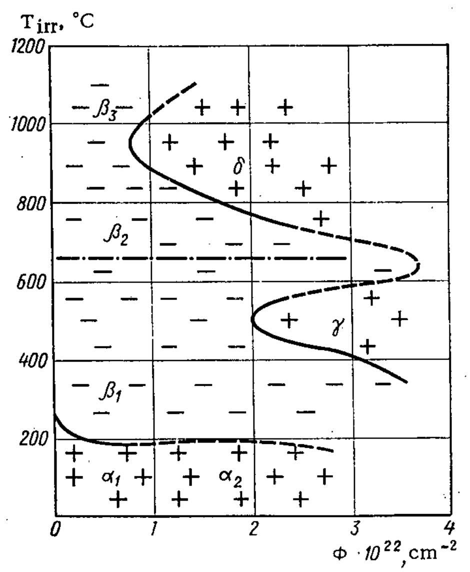

Fig. 1. Generalized diagram of radiation-induced changes in the volume of graphite. The plus and minus symbols denote the swelling and shrinkage regions, respectively.

Generalized Diagram of Radiation-Induced Change in the Graphite Volume. For determining the mechanism of radiation damage to graphite, it is necessary to consider changes in its specific volume due to irradiation and heat treatment in a wide temperature range. On the basis of available data [5-10], including our data [1, 10], we can plot a generalized integral flux vs irradiation temperature diagram, on which the regions where graphite swells or shrinks are indicated (Fig. 1). In view of the complexity and variation of the behavior of various sorts of graphite, the diagram does not have a completely quantitative character. This is due to the fact that the phenomena have not been sufficiently investigated throughout the range of doses and temperatures in question. However, the diagram makes it possible to analyze the overall pattern.

The region α (low-temperature irradiation) is characterized by swelling, which is a consequence of the cumulative implantation and vacancy radiation defects in the crystal. Implantations cause an increase in the interplanar spacing $c/2$, while vacancies cause

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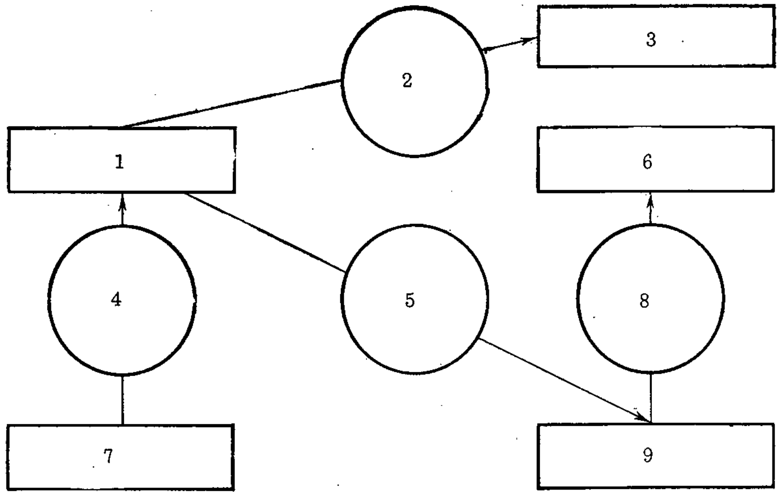

Fig. 2. Diagram of the direction of structural changes in the various processes: 1) standard artificial polycrystalline graphite; 2) high-temperature irradiation (1000°C); 3) radiation-graphitized material; 4) standard graphitization at 2400°C; 5) low-temperature irradiation (100°C); 6) material with restored and improved standard structure; 7) initial imperfect carbon material; 8) annealing at 1500°C; 9) material rendered amorphous by radiation.

a reduction in the mean parameter a. The first factor predominates in the α region. At these temperatures, defects consisting of single vacancies or their groups in the form of implanted C_2 molecules or their groups $(C_2)_2$, $(C_2)_4$, etc., are considered to be stable, since single implanted carbon atoms possess high mobility. The structure of irradiated graphite in the α region can be considered as a peculiar metastable solid solution of implanted carbon atoms in graphite in the presence of a large number of vacancies [1]. Such a structure cannot be obtained by using nonradiation means. Its metastability manifests itself in the fact that regression occurs in heating irradiated graphite to a temperature somewhat exceeding the irradiation temperature. It continues with the rise in the annealing temperature and results in the recovery of the initial structure. The defects vanish by annihilation or by reaching drains (the activation energy in the 150-200°C range is 1.2 eV [12]). However, the structures are somewhat different in the left-hand (α_1) and right-hand (α_2) sides of the region α, and the recovery process occurs in different ways. On the right-hand side α_2, i.e., for large neutron fluxes, the density of defects is so high that the diffraction pattern lines vanish completely, while the recovery of dimensions in annealing starts only at high temperatures (above 600°C).

The region of states β is achieved by irradiation at elevated temperatures; it is characterized by shrinkage. This region has several parts. The bottom part β_1 (up to 600°C) is characterized by the mobility of fine implantation groups. It is considered [13, 14] that the implantations move toward drains and combine into large, less mobile groups, which, the larger they are, the less they affect the volume (in calculation based on a single implanted atom). The accumulation of large groups progresses slowly, and, therefore, the effect of vacancies predominates initially, producing a shrinkage. The implantation effect leading to swelling appears only in prolonged irradiation. This is the region γ, which has the same structure as the region α, but also large interplanar groups. The upper part of the region β (around 600°C and above) is characterized by the mobility of vacancies (the activation energy is 2.6 eV [15]). They combine into continuous chains, which flatten, so that the initial lattice constant is preserved as much as possible. A reduction in the volume corresponds to a concentration of vacancies, the more so, the larger the group. Microporosity, which leads to swelling, develops apparently as a result of prolonged irradiation. This is the region δ. At high temperatures, the process is accompanied by a large amount of shrinkage and leads more rapidly to the δ region.

Some Recent Data. We investigated artificial polycrystalline graphite with different structures: graphite produced from calcined petroleum coke with a coal pitch binder according to the standard technology (with graphitization at 2300°C) and graphite produced according to a somewhat different technology

– with an addition of not readily graphitizable carbon black. The graphite specimens were irradiated in the SM-2 reactor with a flux of $3 \cdot 10^{14}$ neutrons/cm^2 . sec (E $\geq$ 0.18 MeV) at temperatures of 100, 170, 600, and 900°C and a flux of $(0.8\text{-}1.2) \cdot 10^{22}$ neutrons/cm^2.

The following was established as a result of measuring the dimensions of specimens and x-ray structural analysis [10]:*

At low irradiation temperatures, both graphite specimens – the standard and the nonstandard specimens – experience swelling, while the lines on diffraction patterns disappear completely; in the annealing of these graphites, the reversible shrinkage and the appearance of diffraction lines occur only at high temperatures (~1000°C), and a shrinkage of specimens to dimensions smaller than the initial ones is observed after annealing to 1500°C. The length of unirradiated control specimens does not change. After annealing, the diffraction patterns of the irradiated specimens indicate a more perfect structure; after high-temperature irradiation (600-900°C), both graphite specimens shrink in size, while the structure of nonstandard graphite containing the not readily graphitizable component (carbon black) becomes more perfect.

Analyzing the proposed diagram and the additional data, we arrive at the conclusion that a mechanism which can be termed radiative graphitization, which is connected with the presence of highly mobile defects produced by irradiation, acts in the top part of the β region (β_3). Radiative graphitization consists in improving the degree of perfection of the structure of artificial graphite, which has technological imperfections. In particular, the "frozen" microstresses can be released, the composition of disoriented graphite lattice sections in crystallites can be improved, and the structure of intercrystallite boundaries can be ordered. It is interesting that this process of more thorough graphitization occurs to a certain extent even in the presence of a not readily graphitizable component (carbon black) in the structure.

Ordering improves the composition of the structure, and, in the final analysis, leads to graphite shrinkage, which is observed in the region β_3. This is apparently indicated by shrinkage to values below the initial dimensions of specimens irradiated at a low temperature. In activation by annealing at high temperatures, the stored frozen radiation defects exert once (although to a lesser degree) the same action as the generated defects if the irradiation is performed at high temperatures. Comparison between these processes makes it possible to consider high-temperature irradiation (in the top part of the region β) as continuous annealing in the presence of mobile radiation defects. In particular cases, this process can be utilized for technological purposes. The flowsheet of the processes is shown schematically in Fig. 2.

The ordering process resulting from radiative graphitization is opposed by the disordering process of radiative amorphization if the irradiation is performed at a low temperature. On the whole, this results in a phenomenon characteristic for order – disorder systems, which is connected with the presence of two possible structural phases of black carbon: crystalline graphite and amorphous carbon black. Irradiation can initiate transition toward the phase which is stable at a given temperature. The metastable disordered phase is fixed at low temperatures (in the region α). At high irradiation temperatures (in the region β_3), even a not readily graphitizable material with a low degree of structural perfection shows a tendency to improve it. The best way to check this is to perform high-temperature irradiation of an imperfect, not readily graphitizable raw material that has been subjected to high-temperature heat treatment. Preliminary low-temperature irradiation of such raw materials for the accumulation of activating radiation defects that can promote graphitization is also of interest.

It should be mentioned that, during prolonged irradiation at a high temperature, the transition to the δ region is probably connected with the development of shrinkage microstresses and microcracks, which progress as a result of the generated vacancies. This assumption and the above assumption concerning vacancy pores have to be checked in microstructure investigations.

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*In collaboration with Yu. S. Virgil'ev, I. P. Kalyagina, and T. N. Shurshakova.

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LIST OF REPORTS SUBMITTED TO THE SECOND PANEL OF THE COMECON '70 CONFERENCE

Redesigning of the VVR-S Type Research Reactor, and Operating Experience – F. Szabó, L. Varkony, and L. Frankel (Hungary)

A brief review is given of the basic stages in the redesigning of the VVR-S type reactor. The nuclear characteristics of the core are cited.

While the reactor was in service, the neutron flux was optimized with attention given to the requirements of research work, isotope production, and economic utilization of fuel assemblies.

High neutron flux with the required neutron spectrum is made available for activation analysis work. The basic solutions contributing to improved operating conditions of the reactor, improved expediting of research work, and expanded production of isotopes, are cited in the article.

Testing Experimental Fuel Assemblies for Power Reactors in the MR Reactor – V. V. Goncharov, E. P. Ryazantsev, L. A. Goncharov, and A. B. Kruglov (USSR)

A procedure for conducting tests of fuel elements and fuel-element materials for power reactors in subloop test rigs is discussed. Concrete examples of the ways in which subloop experiments are set up and carried out are reported.

Brief results of MR in-pile tests of fuel elements consisting of sintered uranium dioxide meat clad in zirconium alloy jackets, for water-cooled water-moderated reactors and for boiling-water reactors designed for nuclear electric power generating stations, are reported.

Validation of the Selection of Zirconium Alloy for Cladding of Fuel Elements in the VVÉR-440 Series-Produced Power Reactors – A. D. Amaev, R. S. Ambartsumyan, V. V. Goncharov, A. M. Glukhov, K. P. Dubrovin, E. G. Ivanov, L. M. Ionova, A. B. Kruglov, L. M. Lebedev, E. P. Ryazantsev, A. V. Sedova, and I. A. Anisimova (USSR)

Results of investigations carried out on the structure, mechanical properties, and corrosion resistance of zirconium alloy with 1% Nb, for use in the fabrication of cladding for fuel elements in VVÉR type reactors, depending on various factors (impurity content, temperature, environment, etc.), which exert a serious affect on the properties of the alloy, are reported. Some relationships are established between the structure of the metal and the properties of oxide films.

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Results of tests carried out on experimental fuel elements clad with Zr + 1% Nb alloy, in the MR reactor, in pressurized water and in boiling water, are presented, as well as data from investigations of the cladding material in a hot materials studies laboratory.

Information is cited on the effect of pile radiation, coolant composition, mechanical stresses, and hydrogen content in the alloy, on changes occurring in the properties of the alloy. It is demonstrated that changes in the strength characteristics and plastic characteristics of the alloy Zr + 1% Nb are affected primarily by radiation-induced hardening, hydrogen content, and the orientation of hydrides present.

Expansion of Experimental Capabilities of the MR Reactor for Subloop Tests on Fuel Elements and Fuel-Element Materials – V. V. Goncharov, E. P. Ryazantsev, P. I. Shavrov, P. M. Egorenkov, E. N. Babulevich, I. M. Novikov, V. K. Fishevskii, A. B. Kruglov, and L. K. Zakharov (USSR)

The Training Reactor of Budapest Polytechnic Institute – D. Csom, I. Diós, L. Frankel, et al. (Hungary)

Rigging and assembly work is now in progress, on the premises of Budapest Polytechnic Institute, on a 10 kW nuclear reactor to be used in the research and training programs of Budapest colleges.

The basic features of this project are reported on. The design and experimental programs worked out at Central Physics Research Institute for optimization of the nuclear-physics parameters of the reactor core are described. Results of the program are cited. A description is given of the five training and research programs being carried out at the reactor and in laboratories.

Reactor Design in the Polish Peoples Republic – (Review Paper)

In 1958, Poland's first research reactor, a VVR-S type reactor delivered by the Soviet Union, was commissioned.

In subsequent years, the power output of this reactor was raised in two stages. In the first stage, the reactor power was raised from 2 to 4 MW in 1964, by changing the design of the fuel assemblies which incorporate type EK-10 fuel elements. This led to a higher surface heat transfer coefficient between the fuel assemblies and the primary-loop coolant, and resulted in lower temperature gradients over the length of the fuel assemblies. Designs of the new fuel assemblies are cited in the article.

In the second stage, the VVR-S reactor output was raised to 8-10 MW in 1966 by changing the design of the VVR-SM type fuel elements, and by resorting to the use of a beryllium reflector. The increased reactor power output stemmed from redesign of the reactor. The design of the new beryllium-reflected reactor core, and the principal components which were rebuilt, are described.

In addition to this VVR-S type reactor, work has begun more recently on the construction of a high-flux research reactor with a core similar to that of the Soviet MR research reactor. The design of the reactor and its equipment are described, and opportunities for utilizing the reactor in materials irradiation, isotope production, and physics research on neutron beams, are indicated.

The SM-2 High-Flux Reactor (Operational and Technological Characteristics) – V. A. Tsykanov, P. G. Aver'yanov, M. D.

Declassified in Part - Sanitized Copy Approved for Release 2013/02/25 : CIA-RDP10-02196R000300080002-2

Aleshin, Yu. P. Kormushkin, R. I. Korotkov, A. S. Kusovnikov, B. V. Samsonov, and G. P. Lobanov (USSR)

The physical, technological, and performance characteristics of the SM-2 reactor achieved following redesign and modernization of the core, the process loops, and experimental devices, are presented. Technological characteristics of the performance of the reactor loops and systems are reported. It is demonstrated that the optimum reactor campaign and operational schedule were selected for a large number of experimental devices operating simultaneously.

The basic engineering costs indices of the performance of the SM-2 reactor such as reactor utilization factor, percentage U^{235} burnup in the fuel assemblies loaded into the core, and the number of specimens being irradiated, are cited.

Materials Testing Methods in the SM-2 High-Flux Reactor – V. A. Tsykanov and B. V. Samsonov (USSR)

Methods for investigating materials in the SM-2 high-flux reactor are presented. Special features of the reactor which have some bearing on the procedures and organization of research are pointed out. Specific examples of subloop and capsule type irradiation devices and devices for in-pile investigations (mechanical properties of structural materials, behavior of fuel materials, electrical engineering parameters of insulating materials, precision in thermocouple readings) are cited.

Experience in Improving Reactor Control and Monitoring Equipment in the VVR-S(M) Type Research Reactor at the Central Physics Research Institute – D. Pallagy (Hungary)

Experience in the operation of control and monitoring systems in operation over a twelve-year period, and improvements and additions carried out during that time, are described. The systems operate reliably, carrying out their functions completely; they do not have to be replaced in the switchover to computerized controls.

Some Physical Features of the SM-2 Reactor, and Comparison of the SM-2 with Other High-Flux Reactors – V. A. Tsykanov, Yu. P. Kormushkin, and R. I. Korotkov (USSR)

The basic physical features of the SM-2 research reactor which are responsible for some of the technical advantages belonging to reactors of this type are cited. A procedure for comparing high-flux reactors, with their productivity and cost factors taken into account, is proposed. This procedure provides a basis for comparing trap type reactors (SM-2 and HFIR reactors), and trapless high-flux reactors (Savannah River reactor), with the inference that trap type reactors, with their appreciable technical advantages, may by quite the equals of large reactors of the type built at the Savannah River nuclear power plant in terms of productivity.

Experimental Determination of Radiation Energy Release Rate in Some Structural Materials in the VVR-S Reactor of the Institute of Nuclear Physics of the Academy of Sciences of the Uzbek SSR

– R. S. Abdullaev, K. P. Vlasov, M. S. Pirogov, Yu. V. Polyak, Yu. S. Skripnikov, N. T. Sychev, V. S. Terekhov, and D. K. Shiryaev (USSR)

Investigations were carried out in the VVR-S reactor of the Institute of Nuclear Physics of the Academy of Sciences of the Uzbek SSR, testing specimens of the materials Kh18N9T, STZ, Rv, aluminum, AMG, zirconium, and molybdenum. Results obtained through the use of different methods to determine the rate of radiation energy production are compared. The dependence of the rate of radiation energy release on the reactor power level, and the effect of a cadmium shield on the amount of energy released, are discussed. The energy distribution over the height of a 5/10 cell is compared to the fast flux distribution and thermal flux distribution.

A computational formula for the distribution of radiation energy release over the height of the 5/10 cell of the VVR-S reactor is derived.

Investigations of Reactor Materials Conducted at the Nuclear Research Institute (Poland)

Development and Investigation of Pearlitic Steels for the Steam Generators of Nuclear Power Plants with Sodium-Cooled Reactors – G. I. Kapyrin, I. V. Gorynin, N. N. Gribov, Yu. F. Balandin, S. S. Shurakov, and Yu. M. Trapeznikov (USSR)

Estimates of the Serviceability of Austenitic Steels for Nuclear Power Plants with Sodium-Cooled Reactors – G. I. Kapyrin, N. N. Gribov, Yu. F. Balandin, S. S. Shurakov, and T. I. Ivanova (USSR)

Estimates of the Serviceability of Cladding in Fuel Elements for Fast Reactors – Yu. I. Likhachev and A. A. Proshkin (USSR)

Long-Term Strength and Creep Behavior of Nickel Alloy Irradiated at Different Temperatures – S. N. Votinov, V. K. Shamardin, and V. I. Prokhorov (USSR)

The effect of neutron irradiation on long-term strength and creep in the alloy Kh20N78T at the temperature 650°C, with tests conducted for periods less than 1000 h, is studied. The irradiation was carried out in the SM-2 reactor, with an integrated flux of $5 \cdot 10^2$ neutrons/cm^2 ($E \geq 0.85$ MeV), at temperatures of 500 and 780°C. It was found that the fall-off in 100 h long-term strength attained its maximum after irradiation at 500°C. The post-exposure microstructure of the alloy after the tests is characterized by the presence of porosity along the grain boundaries.

Effect of Neutron Irradiation on the High-Temperature Properties of Type Kh13 Stainless Steels – S. N. Votinov, V. D. Balashov, V. I. Zinkovskii, E. A. Krylov, V. I. Prokhorov, and V. K. Shamardin (USSR)

The mechanical properties of steels type Kh13, of the ferrite – martensite class, with alloying additions of Si, Mo, Nb, B, and so forth, were studied. Coupons of these steels were irradiated by fast neutrons in the SM-2 reactor at the temperature 100°C, at integrated doses of $4 \cdot 10^{20}$, $4 \cdot 10^{21}$, and $4 \cdot 10^{22}$ neutrons/cm^2, and also at T = 350°C up to integrated doses of $8 \cdot 10^{20}$ neutrons/cm^2, and at T = 650° to 700°C at integrated doses up to $2 \cdot 10^{20}$ neutrons/cm^2.

It is shown that softening greater than that characterizing the unirradiated state is detected in all the steels investigated, following exposure to the fast flux, and starting with a certain temperature which depends on the degree of alloying, the integrated dose, and the irradiation temperature. The temperature range for softening of irradiated steels extends from 500° to 650°C.

It is demonstrated that high-temperature ductility in chromium steels deteriorates after neutron irradiation.

Heat-Resistant Steel for Pressure Vessels of Pressurized-Water (Water-Cooled and Water-Moderated) Reactors – I. V. Gorynin, Yu. F. Balandin, Yu. I. Zvezdin, I. A. Razov, and S. S. Shurakov (USSR)

A special class of heat-resistant steels has been developed, on the basis of an extensive research and development program, for the pressure vessels of water-cooled water-moderated reactors using pressurized water. The composition of these steels is: carbon 0.11 to 0.21%, chromium 2.0 to 3.0%, molybdenum 0.6 to 0.8%, vanadium 0.25 to 0.35%. These steels respond well to manufacturing processes and can be used in the fabrication of large welded pressure vessels for nuclear reactors.

The effect of heat treatment conditions on strength and plasticity under short-term tensile load, and proclivity to brittle fracture, were investigated. A study was also made of thermal embrittlement of these steels in long-term service at temperatures in the vicinity of 350°C.

The steels developed in this program show more or less the same resistance to short-term fatigue as that shown by low-alloy and medium-alloy pearlitic steels.

Special attention was given, in the course of developing these steels, to their stability to radiation, and to possible hydrogenation through the interation with water in the event of disruption of the anticorrosion surfacing. Under irradiation with a flux of $\sim 10^{20}$ neutrons/cm^2 ($E \geq 1$ MeV) at temperatures near 300°C, the critical temperature for the brittle-to-ductile transition shifted by 20-50°C in these steels, whereas the shift is greater than 100°C in the case of steels of similar function in service in the USA (grades A302, A212, A516, A350).

A theoretical estimate was made of the maximum possible hydrogen saturation of the steel. Limiting pressures of the hydrogen released when the iron interacts with water at temperatures from 300° to 1000°K were calculated.

Possible hydrogen concentrations in the steel were determined on the basis of the data so obtained. The hydrogen saturation features of the steel during corrosion processes are examined.

Investigation of the Behavior of Zirconium Alloys under Irradiation, and Investigation of Coolant for the VK-50 Boiling-Water Reactor

– V. S. Belokopytov, S. N. Votinov, A. B. Andreeva, M. D. Deribizov, V. K. Shamardin, and P. P. Grinchuk (USSR)

Results of an investigation of the alloys Zr + 1% Ne and Zr + 2.5% Nb on specimens irradiated for 13,000 h in the coolant used in the VK-50 boiling-water reactor are reported.

The experimental data obtained on the corrosion behavior of the alloys, and results of mechanical tests, are then compared to data on autoclave tests and reactor subloop tests of the zirconium alloys.

An estimate of the service life of the alloys Zr + 1% Nb and Zr + 2.5% Nb under real conditions in the VK-50 boiling-water reactor was made on the basis of analysis of the results obtained.

Effect of Isochronous Annealing on the Mechanical Properties and Morphology of Hydride Inclusions in Irradiated Zr + 1% Nb Alloy – E. G. Ivanov, L. A. Elesin, L. M. Lebedev, and V. V. Chizhaev (USSR)

Results of investigations of coupons of an irradiated central tube in a fuel assembly of a reactor in the Novaya Voronezh' Nuclear Power Station, made with isochronous annealing at 300-800°C, are reported. The following techniques were applied in the investigations: optical metallography, electron transmission microscopy, studies of mechanical properties, microhardness measurements, determination of hydrogen present.

A central tube 12.0 × 0.85 mm across was irradiated in a reactor fuel assembly at t = 250-270°C for 15,000 h with an integrated neutron flux ($E \geq 1$ MeV) of about $2 \cdot 10^{21}$ neutrons/cm^2. During the irradiation, the tube was oxidized on both sides, resulting in the formation of a uniform oxide film up to 0.03 mm thick, while the hydrogen content attained the level of 0.08 wt. %.

It was found that considerable annealing out of defects formed during irradiation takes place at the temperature 450°C, and that even though the hydrogen content shows no signs of diminishing, ductility rises from zero to 10%. Further annealing brings to light a complex relationship between changes in ductility accompanied by changes in the morphology of the hydride inclusions, a decrease in the amount of hydrogen present in the material, and dissolution of oxygen from the film in the alloy.

Production of Zirconium by Electrolysis of Fused Salts – G. Mândru (Rumania)

Investigations were conducted on a laboratory facility with a capacity of 40 to 50 g zirconium per cycle, with electrolysis of a melt of K_2ZrF_6 and NaCl in an argon environment. The composition of the zirconate, percentage figures, was: Al 10^{-1}, Si 10^{-1} to 10^{-2}, Fe > 10^{-2}, Mn 10^{-2} to 10^{-3}, Mg ≥ 10^{-2}, Pb 10^{-3}, Ti 10^{-3}, Cu 10^{-3} to 10^{-4}, Ca > 10^{-2}, Hf > 10^{-1}. The cathode was a stainless steel rod, while the anode was a graphite crucible.

The process variables were: voltage 4-5 V, initial current density at cathode 300 to 500 A/dm^2, temperature 800-820°C, duration 1.5 to 3 h.

A mixture of zirconium and the salts NaF, KF, K_2ZrF_6 was deposited on the cathode during the electrolysis. The salts were removed by washing with distilled water, followed by acidification with HCl.

Over 75% of the resulting zirconium powder consisted of particles in the 40-150 mm size range. The chemical composition of the resulting metal, in percentage figures, was: Fe > 10^{-1}, Ni 10^{-2}, Cr 10^{-1}, Si ≥ 10^{-1}, Ti 10^{-2} to 10^{-3}, Mg 10^{-3}, Al 10^{-2}, Cu 10^{-3}, Mn 10^{-1}, Hf 10^{-1} to 10^{-2}. The elements Fe, Cr, Mn, and Si get into the zirconium through corrosion of the process equipment.

Separation of the hafnium from the zirconium took place in the course of the electrolysis, with a degree of purification 95-97%.

Process variables optimizing the yield of metal were arrived at. It is reported that repeated utilization of the electrolyte bath is advisable only when less than 80% of the metal is extracted in the first cycle.

Investigation of the Radiation Stability of Pressure Vessel Material for a Heavy-Water Reactor – S. N. Votinov, V. A. Tsykanov, B. V. Sharov, V. K. Shamardin, V. I. Prokhorov, G. P. Lobanov, V. N. Shulimov, V. S. Yarkovoi, G. A. Sernyaev, V. M. Raetskii, B. V. Samsonov, and A. S. Pokrovskii (USSR)

Irradiation procedures are described. Results of a study of the mechanical properties of the aluminum alloy SAV-1 in the state of stabilizing heat treatment (quench – natural ageing – anneal at 230°C for 20 h) at testing temperatures 20° and 80°C, following irradiation in the SM-2 reactor, are cited. The irradiation was conducted at the temperatures: 50-60°C and 110-130°C in water, 80-110°C in helium, 160-170°C in inert filler gas. The integrated neutron doses were $3.1 \cdot 10^{19}$ to $1.6 \cdot 10^{21}$ neutrons/cm^2. It was shown that changes in the strength characteristics and ductility characteristics of the alloy SAV-1 in the stabilized state, during irradiation at 80-130°C with integrated doses to $1.6 \cdot 10^{21}$ neutrons/cm^2, place no restrictions on the use of the alloy under the physical conditions prevailing in the operation of the heavy-water reactor.

Validation of the Selection of Zirconium Alloy for Cladding of Fuel Elements in Series-Fabricated VVÉR-440 Power Reactors – R. S. Ambartsumyan et al. (USSR)

Studies of Porosity in Cyclotron-Irradiated Steels – Yu. N. Sokurskii, N. P. Agapova, V. D. Onufriev, and F. P. Butra (USSR)

The method of transmission electron microscopy was used to follow out the initial stages of nucleation and development of helium porosity in specimens of OKh16N15M3B steel and OKh20N80 alloy irradiated by helium ions on a cyclotron, to concentrations of 0.012 wt. % helium, with anneals and high-temperature deformation.

Data were obtained on the number of pores per cm^3 of metal, on the growth kinetics of pores, on the size distribution of pores, and also on the relation of pores to dislocations, grain boundaries, and precipitations at different stages of annealing. An estimate was made of the diffusion coefficient of helium atoms in the materials investigated. The kinetics of helium liberation from a solid solution based on a metallic matrix into pores was studied.

A metallographic investigation was conducted of later stages of helium porosity. Structural investigations of helium-saturated specimens were sized up against data from mechanical testing at temperatures to 800°C, i.e., at temperatures at which high-temperature radiation embrittlement is observed.

The development of porosity is observed to speed up during deformation. The embrittlement effect is connected with blockage of intragrain slip, because of pinning of dislocations by fine pores, and with suppression of diffusion creep of the matrix on account of helium atoms, as well as with weakening of grain boundaries because of pile-up of helium pores on grain boundaries.

Effect of Radiation-Induced Changes in Structure on the Properties of Sintered Materials – E. Labusca (Rumania)

Several aspects of the effect of the structure of sintered materials on their behavior under neutron irradiation are discussed.

Selection of process variables is used to produce sintered materials (Al_2O_3 and ferrite base) of identical chemical composition but with different structures and properties.

The structural modifications were revealed by electron microscopy. A relationship with the initial structure and with the neutron flux was found, as well as with changes in the mechanical properties of the Al_2O_3 and in the magnetic properties of the ferrites.

Attempts were undertaken to simulate their radiation effect.

Note on the Distribution of Fission Fragments and Oxygen in Uranium Dioxide – V. P. Vyskubov, V. M. Kosenkov, I. G. Lebedev, and A. S. Pokrovskii (USSR)

X-ray structural analysis was applied to investigation of the state of the UO_2 crystal lattice following high-temperature irradiation of rod-type fuel elements to burnup levels of 9 wt. % uranium. It is proposed, on the basis of data from analysis of the ratio of structural factors of diffraction lines with an even and an odd half-sum of interference indices, that a portion of the fission fragments becomes lodged interstitially in octahedral voids in the UO_2 lattice.

The fact of nonmonotonic variation in the crystal lattice parameters, and widths of diffraction lines over the radius of the fuel element is established. The results are discussed in relation to the distribution of fission fragments and of oxygen in the UO_2 lattice. Analysis of the oxygen ion diffusion equation with respect to the cross section of the fuel element confirms the existence of an extremum point in the oxygen distribution over the radius of the fuel-element meat core.

Investigation of Promising Uranium Carbide and Plutonium Carbide Fuel Compositions for Fast Reactors – I. S. Golovnin, T. S. Men'shikova, F. G. Reshetnikov, B. D. Rogozkin, V. V. Artamonov, R. B. Kotel'nikov, V. S. Belevantsev, S. N. Bashlykov, Yu. K. Bibilashvili, V. S. Mukhin, G. V. Kalashnik, I. G. Lebedev, V. I. Kuz'min, A. S. Piskun, L. K. Druzhinin, and G. V. Titov (USSR)

Investigation of UO_2 Low-Temperature Sintering of UO_2 – I. Mirion and G. Daschelu (Rumania)

Some results of investigations into the effect of process variables on the properties of sintered UO_2 are reported.

Dimensional Stability of Rod Type Fuel Element under Irradiation – V. E. Ivanov and V. F. Zelenskii (USSR)

The effect of various factors, such as radiation growth, swelling, and creep under irradiation on the life span of a rod type uranium fuel element is analyzed. The discussion takes up the operating conditions of the fuel element in the KS-150 reactor. The conclusions drawn from this analysis are compared with results established experimentally in radiation tests on rod type fuel elements. Some inferences are drawn on the prospects for using fuel elements of this type in reactors at burnup levels to 1.5-2%.

Investigation of Working Fuel Bundle (With Carbide Fuel) in the BR-5 Fast Reactor – A. I. Leipunskii, Yu. A. Aleksandrov, O. D. Kazachkovskii, S. N. Votinov, I. G. Lebedev, K. S. Piskun, E. F. Davydov, et al. (USSR)

Some Problems in the Development of Fuel Elements for Fast Reactors – I. S. Golovnin, Yu. K. Bibilashvili, T. S. Men'shikova, A. V. Medvedev, and G. V. Kalashnik (USSR)

Post-Pile Investigations of Fuel Elements in Service for 626 Effective Days in the Core of the VVÉR-1 Reactor of the Novaya Voronezh' Nuclear Power Station – V. S. Belokopytov, S. N. Votikov, Z. I. Pakhomov, M. D. Deribizov, V. Ya. Gabeskiriya, V. F. Portnov, and P. P. Grinchuk (USSR)

Design and Basic Characteristics of the Fuel Element for the VVÉR-1000 Reactor – A. S. Zaimovskii, V. V. Kalashnikov, V. N. Kostrov, L. L. Malanchenko, A. V. Nikulina, and V. S. Yamnikov (USSR)

Investigation of Fuel Elements in a Fuel Assembly in Service for 17,000 h in the Core of the VK-50 Reactor – V. S. Belokopytov, S. N. Votinov, V. M. Sarychev, M. A. Dem'yanovich, B. P. Petin, A. B. Andreeva, M. D. Deribizov, and V. A. Shchepetil'nikov (USSR)

Development of the Technology of Uranium Metal Based Rod Type Fuel Elements for Use in the Production of Fuel Elements for a Critical Assembly in the Czechoslovak A-2 Heavy-Water Reactor Project – E. Vaclavik and S. Kain (CSSR)

A review is presented of papers devoted to the development of the technology of rod type fuel elements at a research and development plant of the uranium industry at Mniška pod Brdy.

Various methods for producing low-alloy uranium alloys of the type U–Fe–Al–Si, and their effect on the uniformity of silicon distribution, are discussed. The technology of extruding uranium rods in the upper range of α-phase temperatures is described. Uranium plastic flow during extrusion is investigated, as well as the effect of fused salts, lubricants, and superficial flaws in the stock, on the quality of the rod products.

Results of measurements of the temperature, pressure, and Poisson behavior in extrusion are reported, and an explanation is offered for the $\alpha-\beta$-transformation occurring. The effect of different sets of heat treatment conditions to which the uranium rods are subjected on the structure and growth coefficients, and also on the strength and soundness of U–Ti welded joints, is discussed. Production of tubing for cladding, and the cladding technology, are discussed, as well as the effect of several factors on the strength of the clad – fuel bond and clad – titanium plug bond.

This technology was applied to the fabrication of 40 tons of fuel elements for the critical assembly in the A-2 reactor.

<u>Investigation of Rod Type Fuel Elements for the Heavy-Water Gas-Cooled KS-150 Reactor</u> – Yu. M. Golovchenko, M. A. Vorob'ev, B. A. Bychkov, A. S. Davidenko, V. S. Votinova, V. T. Deripasko, A. Ya. Zavgorodnii, A. P. Sych, et al. (USSR)

<u>Calorimetric Method for Determining Burnup of A-1 Reactor Power Station Fuel Elements While in Service</u> – S. Novák, M. Budinsky, and J. Bereš (CSSR)

A method for measuring the degree of burnup in separate fuel elements of the A-1 nuclear power station while the elements are in service is reported on. The amount of heat generated in a single fuel element is determined, with allowance for all possible losses, by measuring (indirectly) the amount of heat carried off by the gas. The relationship between the change in the isotope composition of the fuel and the amount of heat given off is determined. The scheme of the computation program is presented, with a discussion of the accuracy attainable by the method.

<u>Study of Migration of Fission Fragments in Fuel Elements, by the γ-Spectrometry Method</u> – A. V. Sukhikh and E. F. Davydov (USSR)

Results of measurements of the concentrations of the fission fragments Cs^{137}, Ce^{144}, Zr^{95}, Nb^{95}, and Ru^{106} as they vary with the height of experimental ceramic fuel base fuel elements, differing in the type of fuel, fuel density, duration and temperature of irradiation, number of cycles of variation in reactor output power, manner of fuel packing in the element volume, are reported. An analysis of the redistribution of the fission fragments as a function of those parameters is also made. The dependence of the distribution of daughters of fission products on the chemical nature of the parent fragments and on the thermodynamic properties of the latter is discussed. Anomalies in the behavior of poorly volatile fragments are explained in terms of the formation and decay of their compounds.

<u>Utilization of Boron Carbide in Nuclear Engineering</u> – F. Constantinescu (Rumania)

Conventional methods for production of B_4C of high purity and stoichiometric composition run into severe difficulties. A method is developed here for producing high-purity B_4C at low temperatures (1450-1500°C). Grain growth in polycrystalline B_4C and in single crystals is investigated.

Effect of Alloying with Elements Belonging to Groups IVA and VIIA of the Periodic Table on the Radiation Stability of Boron Carbide – V. P. Gol'tsev, T. M. Guseva, V. V. Chesanov, V. K. Nevorotin, and V. B. Ponomarenko (USSR)

Results of an investigation of the effect of alloying with elements belonging to groups IVA and VIIA of the periodic acid on the radiation stability of boron carbide are reported. Compositions with a natural boron carbide base (B_4C + Fe, B_4C + Ti, B_4C + Zr) with 1-2% additions of alloying elements were irradiated in the SM-2 reactor with a thermal flux of $5 \cdot 10^{14}$ neutrons/$cm^2 \cdot$ sec and fast flux of 10^4 neutrons/$cm^2 \cdot$ sec. After irradiation, these compositions were subjected to the following investigations: determination of density and structural stability, study of phase composition, study of changes in the linear expansion coefficient, and also study of the process of liberation of gas with subsequent investigation of the structure in isothermal anneals.

On the basis of the results of these studies, it was found that alloying with elements of the IVA group does more good than alloying with elements belonging to the VIIA group. When a study was made of structural changes following isothermal anneals, some slight susceptibility to radiation damage on the part of B_4C + Fe compositions after isothermal anneals was detected. The favorable effect of these alloying elements on enhancing the radiation stability of the carbide was inferred from the studies.

Effect of Tungsten and Molybdenum on the Radiation Stability of Boron Chromium – Nickel – Iron Alloy – I. Ya. Emel'yanov, R. V. Grebennikov, B. S. Sergeev, and R. K. Pereverzeva (USSR)

Boron chromium – nickel – iron alloys are used as absorbers in the control rods of nuclear reactors.

Widespread acceptance and use of this inexpensive and easy-to-work material is hampered by the formation of helium when B^{10} is irradiated. Buildup of helium in micropores at high irradiation temperatures leads to swelling of parts made from this alloy, to the formation of microcracks and loss of strength. The microstructure of these alloys is two-phase, consisting in the cast form of chromium – nickel – iron base and a brittle acicular boride phase with coarse Me_2B type borides.

Radiation stability of type Kh20N35R2 boron alloys is enhanced appreciably if the boride phase is comminuted and the strength of the metal substrate is improved.

The boride phase can be sized-reduced by hot deformation of the alloy. At irradiation temperatures of 500-600°C, rolled alloys exhibit better radiation stability and greater strength than cast alloys. But these advantages are vitiated when the temperatures are raised. Alloying of boron alloys with tungsten and molybdenum, which improve the refractory behavior of the metal base, simultaneously brings about size reduction of the boride phase. The boride forming then has the same form as M_3B_2 with a hexagonal type VB_2 lattice.

Bending strength or unirradiated chromium – nickel – iron boron alloys alloyed with tungsten and molybdenum is improved over that of unalloyed borno alloys, from 70 to 85-100 kg/mm^2. When irradiated with a flux of $3 \cdot 10^{20}$ neutrons/cm^2 at 750-850°C, the strength of the unalloyed alloy drops to 10 kg/mm^2, while alloy with additions of tungsten and molybdenum shows a rise in strength to 50 kg/mm^2. The density of boron alloys with no additions of tungsten and molybdenum drops by 20-25% under irradiation (at the parameters listed above), but by 15% in the case of alloys with additions of tungsten and molybdenum.

Alloying of boron alloys with tungsten and molybdenum also reduces warping (sag under irradiation) to a considerable extent.

Effect of Irradiation on the Properties of Boron-Containing Absorbing Materials – V. P. Gol'tsev, T. M. Guseva, V. K. Nevorotin, V. B. Ponomarenko, N. V. Krasnoyarov, and S. A. Kuznetsov (USSR)

Results of radiation tests on (n, α)-absorbers (TaB_3, CrB_2, HfB_2, EuB_6, EuB_{16}, LaB_{16}) and their mixtures irradiated in the SM-2 reactor with a thermal flux of $5 \cdot 10^{14}$ neutrons/cm^2 · sec and fast flux of $1 \cdot 10^{14}$ neutrons/cm^2 · sec for 3350 h are reported. Investigation of the radiation stability entails also investigating structural stability and dimensional stability, studies of mechanical properties, dilatometric investigations, and also the study of the process of gas liberation over the temperature range from 200° to 1000°C.

It was found that not all the compositions are capable of withstanding irradiation, because of accumulation of helium, which generates significant stresses in the material. Studies of the density and microstructure made it possible to discern changes in the geometric parameters, and considerable damage to the structure itself was revealed.

The study of the process of gas liberation from these compositions demonstrated their different ability to retain the gas, as well as the presence of different mechanisms by which the gas escapes from parts made from the material. On the basis of the results of these investigations, conclusions were drawn as to the radiation stability of some borno-containing compounds, and recommendations were advanced on their possible use as materials for control rod systems in various nuclear reactors.

Radiation Stability and Effectiveness of Dysprosium Absorber Materials – I. Ya. Emel'yanov and R. V. Grebennikov (USSR)

Dysprosium, upon absorbing neutrons, forms a chain of daughter isotopes with a high cross section for neutron absorption.

Results of a study of the stability of pressed, and later sintered, specimens made from oxides of rare earths with different contents of titanium oxide are cited, since information has been provided on the positive effect of titanium oxide on the stability of other oxides of rare earths.

A procedure for measuring absorptivity on a physical test bed is described, and experimental data on the effect of additions of the oxides of hafnium, gadolinium, erbium, and holmium on the effectiveness of dysprosium oxide are reported.

Results of calculations of burnup of dysprosium with additions of gadolinium, erbium, holmium, and hafnium are also cited. The results of the effect of long-term irradiation ($3.3 \cdot 10^{20}$ neutrons/cm^2 with depression taken into account) in a reactor at temperatures 550-650°C and 750-850°C on the strength and dimensions of sintered ceramic specimens with a dysprosium oixde base are reported. It is shown that dysprosium oxide case ceramic materials exhibit excellent radiation stability.

Bending strength, dimensions, and shape of the specimens underwent virtually no change.

Radiation Stability of Rare Earth Base Absorbing Materials – V. P. Gol'tsev, T. M. Guseva, N. V. Krasnoyarov, S. A. Kuznetsov, V. B. Ponomarenko, and V. K. Nevorotin (USSR)

Results of an investigation of oxides of the rare earths (Eu, Sm, Dy, Gd), and their dispersion in a metallic matrix (Cr, W, Ta, Mo, Ni, Cu) following neutron irradiation are cited.

Specimens were irradiated in the SM-2 reactor at temperatures of 300-500°C by a thermal flux of $5 \cdot 10^{14}$ neutrons/cm^2 · sec and fast flux of 10^{14} neutrons/cm^2 · sec, for ~150 effective days.

On the basis of the radiation stability criteria for (n, γ)-absorbers, a detailed study was made of the effect of irradiation on the dimensional stability, phase stability, and structural stability of the compositions.

The analysis of the results provided a basis for recommendations as to the applicability of the (n, γ)-absorbers investigated as materials for control rod systems in nuclear reactors to be built in nuclear electric power generating stations.

Radiation Damage Mechanism and Service Life of Structural Graphites at High Temperatures and at High Neutron Flux Levels – V. I. Klimenkov, Yu. S. Virgil'ev, V. R. Zolotukhin, G. I. Sochilin, V. G. Dvoretskii, I. P. Kalyagina, and T. N. Shurshakova (USSR)

PROPAGATION OF ELF AND VLF WAVES NEAR THE EARTH

By **Ya. L. Alpert** and **D. S. Fligel**
Institute of the Earth's Magnetism, Ionosphere, and Radio Wave Propagation of the Academy of Sciences of the USSR

Translated from Russian by **James S. Wood**

This volume presents a comprehensive review of current developments in the area of propagation of VLF and ELF electromagnetics. Frequencies covered range from the VLF maximum of approximately 100 kHz to the ELF minimum of about 1 Hz. Propagation in the waveguide formed by the earth's surface and the lower ionosphere is thoroughly discussed. The book also includes an account of the most recent and sometimes controversial research in magnetospheric plasma above the ionosphere as well as discussions of ELF and VLF wave propagation within the ionosphere.

Values of the refractive index and attenuation factor calculated with regard for the contribution of neutral particles are given for the lower ionosphere (D-region), the outer ionosphere (3000 to 6000 km), and the magnetosphere (to 100,000 km). Various theoretical relations derived rigorously and by asymptotic techniques for the field in the waveguide are subjected to analysis, and the theoretical results are compared with the greater body of experimental data found in the literature.

Excellent for the first year graduate student in physics and electrical engineering, this Special Research Report will also prove invaluable for space, radio, plasma, and theoretical physicists, electrical engineers, geophysicists, astronomers and astrophysicists, and research workers in fluid mechanics.

Ya. L. Alpert is a leading Soviet authority on all aspects of radio wave propagation. Alpert and his co-workers have pioneered methods of analysis of waveforms of "atmospherics" and conducted provocative work on use of electromagnetic waves emitted from nuclear explosions.

SPECIAL FEATURES

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